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

March 21, 2016

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS RE: MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS (CAC NOS. MF4760 AND MF4761)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment Nos. 305 and 309 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively. These amendments consist of changes to the Technical Specifications (TSs) and Renewed Facility Operating Licenses (RFOLs) in response to your application dated September 4, 2014,[1] as supplemented by additional letters.[2]

The amendments change the TSs and RFOLs to allow plant operation from the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) domain to plant operation in the expanded MELLLA Plus (MELLLA+) domain under the previously approved extended power uprate conditions of 3,951 megawatts thermal rated core thermal power. The expanded MELLLA+ operating domain increases operating flexibility by allowing control of reactivity at maximum power by changing flow rather than by control rod insertion and withdrawal.

Enclosure 4 transmitted herewith contains sensitive unclassified information When separated from Enclosure 4, this document is decontrolled.

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[1] September 4, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14247A503).

[2] January 29, 2015 (ADAMS Accession No. ML15029A640); February 6, 2015 (ADAMS Accession No. ML15037A502); April 28, 2015 (ADAMS Accession No. ML15118A717); July 6, 2015 (ADAMS Accession No. ML15187A391); September 4, 2015 (ADAMS Accession No. ML15247A088); October 1, 2015 (ADAMS Accession No. ML15274A467); October 26, 2015 (ADAMS Accession No. ML15299A084); and January 15, 2016 (ADAMS Accession No. ML16019A101).

The NRC staff has determined that its safety evaluation (SE) for the subject amendments contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390. Accordingly, the NRC staff has prepared a redacted, publicly available, non-proprietary version of the SE. Both versions of the SE are enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 305 to Renewed DPR-44
2. Amendment No. 309 to Renewed DPR-56
3. Non-Proprietary Safety Evaluation
4. Proprietary Safety Evaluation

cc w/enclosures 1, 2, and 3: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 305
Renewed License No. DPR-44

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), and PSEG Nuclear LLC (the licensees), dated September 4, 2014, as supplemented by letters dated January 29, February 6, April 28, July 6, September 4, October 1, and October 26, 2015, and January 15, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Renewed Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 305, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. In addition, Renewed Facility Operating License No. DPR-44 is amended by the addition of new license condition 2.C(16), "Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration," as indicated in the attachment to this amendment.

4. This license amendment is effective as of its date of issuance and shall be implemented within 1 year.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: March 21, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 305

RENEWED FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3	3
7g	7g

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

Remove	Insert
3.3-3	3.3-3
3.3-6a	3.3-6a
3.3-7	3.3-7
- - -	3.3-7a
3.4-1	3.4-1
3.4-3	3.4-3
5.0-21	5.0-21
5.0-22	5.0-22

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility, and such Class B and Class C low-level radioactive waste as may be produced by the operation of Limerick Generating Station, Units 1 and 2.

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

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the Peach Bottom Atomic Power Station, Unit 2, at steady state reactor core power levels not in excess of 3951 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 305, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Peach Bottom Atomic Power Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 281 and modified by Amendment No. 301.

(4) Fire Protection

The Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report (SER) dated May 23, 1979, and Supplements dated August 14, September 15, October 10 and November 24, 1980, and in the NRC SERs dated September 16, 1993, and August 24, 1994, subject to the following provision:

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

2. Level 1 performance criteria.

3. The methodology for establishing the RSD strain limits used for the Level 1 and Level 2 performance.

(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall include a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the end-to-end B/Us determined at EPU conditions and a comparison of predicted and measured pressures and strains (RMS levels and spectra) on the RSD. The report shall be submitted within 90 days of the completion of EPU power ascension testing for Peach Bottom Unit 2.

(f) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of the steam dryer as described in the inspection guidelines contained in WCAP-17635-P.

(g) 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 within 90 days following startup from each of the first two respective refueling outages.

(h) 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 paragraphs (f) and (g), 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 paragraph (h).

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

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	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
	AND		
	I.2	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power-High scram setpoints defined in the COLR.	12 hours
	AND		
	I.3	Initiate action to submit an OPRM report in accordance with Specification 5.6.8.	Immediately
J. Required Action and associated Completion Time of Condition I not met.	J.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	AND		
	J.2	Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	AND		
	J.3	--------NOTE-------- LCO 3.0.4 is not applicable. --------------------- Restore required channel to OPERABLE.	120 days
K. Required Action and associated Completion Time of Condition J not met.	K.1	Reduce THERMAL POWER to < 18% RTP.	4 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.16	Deleted	
SR 3.3.1.1.17	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.18	Verify the RPS RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.19	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. Wide Range Neutron Monitors					
a. Period-Short	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
b. Inop	2	3	G	SR 3.3.1.1.5 SR 3.3.1.1.17	NA
	5[(a)]	3	H	SR 3.3.1.1.6 SR 3.3.1.1.17	NA
2. Average Power Range Monitors					
a. Neutron Flux-High (Setdown)	2	3[(c)]	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	≤ 15.0% RTP
b. Simulated Thermal Power-High	1	3[(c)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12[(e),(f)]	≤ 0.61 W + 67.1% RTP[(b)(g)] and ≤ 118.0% RTP
c. Neutron Flux-High	1	3[(c)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	≤ 119.7% RTP
d. Inop	1,2	3[(c)]	G	SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	NA
f. OPRM Upscale	≥ 18%[(h)] RTP	3[(c)]	I	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	NA

(continued)

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

(b) 0.55 (W - ΔW) + 61.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

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

(d) Deleted

(e) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(f) The instrument channel set point shall be reset to a value that is within the Leave Alone Zone (LAZ) around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided the as-found tolerance and LAZ apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP methodologies used to determine the as-found tolerance and the LAZ are specified in the Bases associated with the specified function.

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

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

(h) 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+ domain.

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

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

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

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

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

-----------------------------NOTE-----------------------------
Required limit modifications for single recirculation loop operation may be delayed for up to 12 hours after transition from two recirculation loop operation to single recirculation 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+ domain with a single recirculation loop in operation.	B.1	Initiate action to exit the MELLLA+ domain.	Immediately
C. No recirculation loops in operation. <u>OR</u> Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specifications 3.2.2 and 3.3.2.1;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Control Rod Block Instrumentation for Specification 3.3.2.1; and

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

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 document:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version as specified in the COLR).

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

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

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

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

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

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

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

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

   i) NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 1, June 2009

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

5.6.8 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

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 309
Renewed License No. DPR-56

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), and PSEG Nuclear LLC (the licensees), dated September 4, 2014, as supplemented by letters dated January 29, February 6, April 28, July 6, September 4, October 1, and October 26, 2015, and January 15, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Renewed Facility Operating License No. DPR-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 309, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. In addition, Renewed Facility Operating License No. DPR-56 is amended by the addition of new license condition 2.C(16), "Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration," as indicated in the attachment to this amendment.

4. This license amendment is effective as of its date of issuance and shall be implemented within 1 year.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: March 21, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 309

RENEWED FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3	3
7g	7g

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

Remove	Insert
3.3-3	3.3-3
3.3-6a	3.3-6a
3.3-7	3.3-7
- - -	3.3-7a
3.4-1	3.4-1
3.4-3	3.4-3
5.0-21	5.0-21
5.0-22	5.0-22

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility, and such Class B and Class C low-level radioactive waste as may be produced by the operation of Limerick Generating Station, Units 1 and 2.

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

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the Peach Bottom Atomic Power Station, Unit No. 3, at steady state reactor core power levels not in excess of 3951 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 309, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Peach Bottom Atomic Power Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 283 and modified by Amendment No. 304.

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[1]-The Training and Qualification Plan and Safeguards Contingency Plan and Appendices to the Security Plan.

(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall include a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the end-to-end B/Us from Peach Bottom Unit 2 benchmarking at EPU conditions. The report shall be submitted within 90 days of the completion of EPU power ascension testing for Peach Bottom Unit 3.

(f) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of the steam dryer as described in the inspection guidelines contained in WCAP-17635-P.

(g) 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 within 90 days following startup from each of the first two respective refueling outages.

(h) 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 paragraphs (f) and (g), 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 paragraph (h).

(16) <u>Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration</u>

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

3. This renewed license is subject to the following conditions for the protection of the environment:

A. To the extent matters related to thermal discharges are treated therein, operation of Peach Bottom Atomic Power Station, Unit No. 3, will be governed by NPDES Permit No. PA 0009733, as now in effect and as hereafter amended. Questions pertaining to conformance thereto shall be referred to and shall be determined by the NPDES Permit issuing or enforcement authority, as appropriate.

B. In the event of any modification of the NPDES Permit related to thermal discharges or the establishment (or amendment) of alternative effluent limitations established pursuant to Section 316 of the Federal Water Pollution Control Act, the Exelon Generation Company shall inform the NRC and analyze any associated changes in or to the Station, its components, its operation or in the discharge of effluents therefrom. If such change would entail any modification to

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	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
	AND		
	I.2	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power-High scram setpoints defined in the COLR.	12 hours
	AND		
	I.3	Initiate action to submit an OPRM report in accordance with Specification 5.6.8.	Immediately
J. Required Action and associated Completion Time of Condition I not met.	J.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	AND		
	J.2	Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	AND		
	J.3	------ NOTE ---------- LCO 3.0.4 is not applicable. --------------------- Restore required channel to OPERABLE.	120 days
K. Required Action and associated Completion Time of Condition J not met.	K.1	Reduce THERMAL POWER to < 18% RTP.	4 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE			FREQUENCY
SR	3.3.1.1.16	Deleted	
SR	3.3.1.1.17	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR	3.3.1.1.18	Verify the RPS RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program.
SR	3.3.1.1.19	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. Wide Range Neutron Monitors					
a. Period-Short	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
b. Inop	2	3	G	SR 3.3.1.1.5 SR 3.3.1.1.17	NA
	5[(a)]	3	H	SR 3.3.1.1.6 SR 3.3.1.1.17	NA
2. Average Power Range Monitors					
a. Neutron Flux-High (Setdown)	2	3[(c)]	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	≤ 15.0% RTP
b. Simulated Thermal Power-High	1	3[(c)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12[(e),(f)]	≤ 0.61 W + 67.1% RTP[(b)(g)] and ≤ 118.0% RTP
c. Neutron Flux-High	1	3[(c)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	≤ 119.7% RTP
d. Inop	1,2	3[(c)]	G	SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	NA
f. OPRM Upscale	≥18%[(h)] RTP	3[(c)]	I	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	NA

(continued)

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

(b) 0.55 (W - ΔW) + 61.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

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

(d) Deleted

(e) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(f) The instrument channel set point shall be reset to a value that is within the Leave Alone Zone (LAZ) around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided the as-found tolerance and LAZ apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP methodologies used to determine the as-found tolerance and the LAZ are specified in the Bases associated with the specified function.

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

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

(h) 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.

<u>OR</u>

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

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

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

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

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

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

---------------------------NOTE---------------------------
Required limit modifications for single recirculation loop operation may be delayed for up to 12 hours after transition from two recirculation loop operation to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS (continued)

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+ domain with a single recirculation loop in operation.	B.1	Initiate action to exit the MELLLA+ domain.	Immediately
C. No recirculation loops in operation. <u>OR</u> Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

   1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

   2. The Minimum Critical Power Ratio for Specifications 3.2.2 and 3.3.2.1;

   3. The Linear Heat Generation Rate for Specification 3.2.3;

   4. The Control Rod Block Instrumentation for Specification 3.3.2.1; and

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

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 document:

   1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version as specified in the COLR).

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

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

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

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

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

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

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

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

   i) NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 1, June 2009

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

5.6.8 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 OPERABILE status.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 305 AND 309

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-44 AND DPR-56

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

Proprietary information pursuant to
Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390
has been redacted from this document.
Redacted information is identified by blank space enclosed within double brackets
as shown here [[ ]].

Enclosure 3

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APPENDICES

1.0 INTRODUCTION

By application dated September 4, 2014 (Reference 1), as supplemented by letters dated January 29 (Reference 2), February 6 (Reference 3), April 28 (Reference 4), July 6 (Reference 5), September 4 (Reference 6), October 1, (Reference 27), and October 26, 2015 (Reference 38), and January 15, 2016 (Reference 39), Exelon Generation Company, LLC (Exelon, the licensee), submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The LAR proposes changes to the PBAPS, Units 2 and 3, Technical Specifications (TSs) and Renewed Facility Operating Licenses (RFOLs) to allow plant operation from the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) domain to plant operation in the expanded MELLLA Plus (MELLLA+) domain under the previously approved extended power uprate (EPU) conditions of 3951 megawatts thermal (MWt) rated core thermal power. The expanded MELLLA+ operating domain increases operating flexibility by allowing control of reactivity at maximum power by changing flow rather than by control rod insertion and withdrawal.

The supplemental letters dated January 29, February 6, April 28, July 6, September 4, October 1, and October 26, 2015, and January 15, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 2, 2014 (79 FR 71454).

1.1 Background

General Site Information

The PBAPS site is located in Peach Bottom Township, York County, Pennsylvania, on the west bank of the Susquehanna River. The site is located approximately 38 miles north of Baltimore, Maryland; 19 miles southwest of Lancaster, Pennsylvania; and 30 miles southeast of York, Pennsylvania.

PBAPS, Units 2 and 3, are boiling-water reactor (BWR) plants of the BWR/4 design with Mark I containments. PBAPS, Unit 1, was a high temperature, gas-cooled reactor that is permanently shut down and is currently maintained in an operating SAFSTOR decommissioning condition.

Both units began commercial operation in 1974. The RFOLs for Units 2 and 3 expire in 2033 and 2034, respectively.

Previous Power Uprates

The Atomic Energy Commission issued full power operating licenses for PBAPS, Units 2 and 3, on October 25, 1973, and July 2, 1974, respectively. Both units were licensed at an original licensed thermal power (OLTP) level of 3293 MWt.

By Amendment Nos. 198 and 211 (Units 2 and 3, respectively) dated October 18, 1994, and July 18, 1995, the NRC approved an approximate 5 percent stretch power uprate that authorized an increase in the maximum thermal power level from 3293 MWt to 3458 MWt.

By Amendment Nos. 247 and 250 (Units 2 and 3, respectively) dated November 22, 2002, the NRC approved a 1.62 percent measurement uncertainty recapture (MUR) uprate that authorized an increase in the maximum thermal power level from 3458 MWt to 3514 MWt.

By Amendment Nos. 293 and 296 (Units 2 and 3, respectively) dated August 25, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14133A046), the NRC approved a 12.4 percent EPU that authorized an increase in the maximum thermal power level from 3514 MWt to the current licensed thermal power (CLTP) level of 3951 MWt. The EPU power level of 3951 MWt represents an increase of approximately 20 percent above the OLTP level of 3293 MWt.

Proposed MELLLA+ Operation

As discussed in Section 2.1 of Attachment 1 to the licensee's application, operation of BWRs requires that the reactor core reactivity balance be maintained to accommodate fuel burn-up. BWR operators typically have two methods to maintain this reactivity balance: (1) control rod movements or (2) reactor recirculation core flow adjustments. Because of operator ease and the effectiveness and distributed effect of void reactivity feedback through the reactor core, recirculation flow adjustments are the preferred reactivity control method.

The EPU for PBAPS, Units 2 and 3, was implemented by extending the MELLLA operating domain up to EPU rated thermal power (RTP) levels. However, this reduces the available core 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 core flow window and compensate for reactivity loss with control rod movement.

The proposed MELLLA+ amendment would increase the operating boundary to permit PBAPS operation at the CLTP of 3951 MWt, with a core flow as low as 83 percent. The licensee stated that by operating in the MELLLA+ domain, a significantly lesser number of control rod manipulations would be required than is currently required in the present operating domain. The licensee further stated that reducing the number of rod manipulations would represent a significant improvement in operating flexibility, as well as providing for safer plant operation. Specifically, reducing the number of control rod manipulations would reduce the likelihood of events initiated by operator errors by reducing the quantity of reactor maneuvers.

The proposed MELLLA+ core operating domain expansion would not require major plant system modifications. It would primarily involve changes to the operating power/core flow map, changes to a small number of instrument setpoints and application of the detect and suppress solution - confirmation density (DSS-CD) stability solution for the oscillation power range monitor (OPRM). The proposed TS and RFOL changes are discussed below in Section 1.4 of this Safety Evaluation (SE).

1.2 Licensee's Approach

As discussed in Section 1.0 of Attachment 1 to the application dated September 4, 2014 (Reference 1), the PBAPS MELLLA+ LAR is based on the following NRC-approved licensing topical reports (LTRs):

- GE-Hitachi Nuclear Energy (GEH) LTR, NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus" (Reference 7). This LTR is referred to as the "M+ LTR" throughout this SE. The M+ LTR defines the scope of the evaluations required to support operation in the expanded operating domain. The LTR dispositions the applicable technical review areas either by applicability of a generic assessment or by identifying plant-specific analyses required in a licensee's LAR to support MELLLA+ operation.

- GEH LTR, NEDC-33075P-A, Revision 8, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density [DSS-CD]" (Reference 8). This LTR is referred to as the "DSS-CD LTR" throughout this SE.

- GEH LTR, NEDE-33147P-A, Revision 4, "DSS-CD TRACG Application" (Reference 9).

- GEH LTR, NEDC-33173P-A, Revision 4, "Applicability of GE Methods to Expanded Operating Domains" (Reference 10). This LTR is referred to as the "Methods LTR" throughout this SE.

- GE Nuclear Energy (GE) LTR, NEDE-32906P-A, Revision 3, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses" (Reference 11).

- GE LTR, NEDE-32906P, Supplement 1-A, "TRACG Application for Anticipated Transient Without Scram Overpressure Transient Analyses" (Reference 12).

- GEH LTR, NEDE-32906P, Supplement 3-A, Revision 1, "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO [anticipated operational occurrence] and ATWS [anticipated transient without scram] Overpressure Transients" (Reference 13).

Attachment 4 to the licensee's MELLLA+ application dated September 4, 2014, contains GEH report NEDC-33720P, Revision 0, "Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 & 3 Maximum Extended Load Line Limit Analysis Plus." This report, referred to as the "SAR" throughout this SE, provides the technical bases for the LAR and contains an integrated summary of the underlying safety analyses and evaluations performed by GEH, specifically for PBAPS. The SAR provides a disposition of the M+ LTR technical review areas either by confirming the applicability of generic assessments or by providing plant-specific analyses. The SAR also addresses applicable limitations and conditions in the NRC staff's SEs that approved the LTRs cited above. Attachment 4 is a proprietary (i.e., non-public) version of the SAR. A non-proprietary (i.e., public) version of the SAR is contained in Attachment 5 to the MELLLA+ application.

The reactor cores for PBAPS, Units 2 and 3, transitioned from General Electric (GE) GE14 fuel to Global Nuclear Fuel (GNF) GNF2 fuel as part of the implementation of the EPU amendment. The EPU was implemented in fall 2014 for Unit 2 and fall 2015 for Unit 3. The current operating cores for both units contain only GNF2 fuel. No fuel design changes were requested as part of the MELLLA+ LAR. The Methods LTR (referenced above) contains a limitation that it was only applicable to GE14 and earlier GE fuel designs. The use of GE Methods for GNF2 fuel is supported by NRC-approved GEH LTR NEDC-33173, Supplement 3P-A, Revision 1, "Applicability of GE Methods to Expanded Operating Domains - Supplement for GNF2 Fuel" (Reference 14).

1.3 Method of NRC Review

The NRC staff review evaluated the licensee's assessment of the impact of the proposed MELLLA+ on the applicable PBAPS design-basis analyses. The NRC staff reviewed the licensee's application and supplements. The NRC staff also performed audits in relation to the following topics:

- Operator actions during potential anticipated transient without scram (ATWS) and thermal-hydraulic instability events (see SE Section 3.3.10 (SAR Section 10.6)).

- Sensitivity calculations and methodologies for ATWS with instability (ATWSI) events using GEH evaluation model TRACG (see SE Appendix A (SRXB-RAI-18)).

The NRC staff reviewed the LAR to ensure 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.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the LAR, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and conditions placed on the methods. In addition, the NRC staff considered the effects of the changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed MELLLA+ conditions.

The NRC staff performed this review, in part, by using relevant sections of the review guidance in NRC, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001, Revision 0 (Reference 15). Although the MELLLA+ LAR is not an EPU, and RS-001 guidance is not wholly applicable, the NRC staff determined that RS-001 provides a good framework for the review of certain portions of the LAR.

Details of the NRC staff's technical evaluation are provided in Section 3.0 of this SE. The technical evaluation is organized as follows:

- SE Section 3.1 provides an overview of the PBAPS MELLLA+ SAR (i.e., GEH report NEDC-33720P, Revision 0, "Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 & 3 Maximum Extended Load Line Limit Analysis Plus," contained in Attachment 4 to the application dated September 4, 2014).

- SE Section 3.2 discusses topics that were dispositioned generically for PBAPS in the SAR by the licensee in accordance with the M+ LTR.

- SE Section 3.3 discusses topics that were dispositioned on a plant-specific basis for PBAPS in the SAR by the licensee in accordance with the M+ LTR.

- SE Section 3.4 provides NRC staff evaluations of various topics in accordance with the guidance in RS-001.

- SE Section 3.5 provides an NRC staff evaluation of the limitations and conditions associated with the LTRs referenced above in SE Section 1.2.

- SE Section 3.6 provides an NRC staff evaluation regarding use of the TRACG code models for ATWSI events for PBAPS.

SE Section 4.0 provides the NRC staff evaluation of the proposed RFOL and TS changes. The specific RFOL and TS changes requested by the licensee are summarized in SE Section 1.4.

1.4 Proposed Renewed Facility Operating License and Technical Specification Changes

The following is a summary of the proposed RFOL and TS changes associated with the LAR. The proposed changes are shown in Attachment 2 to the application dated September 4, 2014, as supplemented by the licensee's letter dated October 26, 2015. The NRC evaluation of these changes is provided in SE Section 4.0.

1.4.1 License Condition - Feedwater Temperature

The RFOL would be revised to add new License Condition 2.C(16), which would prohibit operation in the MELLLA+ domain with a feedwater heater out of service, resulting in more than a 10 degrees Fahrenheit (°F) reduction in feedwater temperature below the design feedwater temperature.

1.4.2 TS 3.3.1.1 - Reactor Protection System Instrumentation

TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," would be revised to support implementation of the DSS-CD stability solution. The TS would be revised as follows:

- Modify Limiting Condition for Operation (LCO) Conditions I and J.
- Add new LCO Condition K.
- Delete Surveillance Requirement (SR) 3.3.1.1.19.
- Revise TS Table 3.3.1.1-1, Functions 2b and 2f.

1.4.3 TS 3.4.1 - Recirculation Loops Operating

TS 3.4.1, "Recirculation Loops Operating," would be revised to prohibit single loop operation (SLO) in the MELLLA+ domain. The TS would be revised as follows:

- Add a new note to the LCO.
- Designate existing LCO Condition B as Condition C.
- Add new LCO Condition B.

1.4.4 TS 5.6.5 - Core Operating Limits Report

TS 5.6.5, "Core Operating Limits Report (COLR)," would be revised to support implementation of the DSS-CD stability solution. The TS would be revised as follows:

- Revise TS 5.6.5.a.5.

1.4.5 TS 5.6.8 - Oscillation Power Range Monitoring Report

New TS 5.6.8, "OPRM Report," would be added to support implementation of the DSS-CD stability solution.

2.0 REGULATORY EVALUATION

General Design Criteria

The construction permit for PBAPS, Units 2 and 3, was issued by the Atomic Energy Commission (AEC) on January 31, 1968. As discussed in Appendix H to the PBAPS Updated Final Safety Analysis Report (UFSAR), during the construction/licensing process, both units were evaluated against the then-current AEC draft of the 27 General Design Criteria (GDC) issued in November 1965. On July 11, 1967, the AEC published, for public comment in the *Federal Register* (32 FR 10213), a revised and expanded set of 70 draft GDC (hereinafter referred to as the "draft GDC"). Appendix H of the PBAPS UFSAR contains an evaluation of the design basis of PBAPS, Units 2 and 3, against the draft GDC. The licensee concluded that PBAPS, Units 2 and 3, conform to the intent of the draft GDC.

On February 20, 1971, the AEC published in the *Federal Register* (36 FR 3255) a final rule that added Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants" (hereinafter referred to as the "final GDC"). Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the NRC's Staff Requirements Memorandum for SECY-92-223, dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. At the time of the 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.

The licensee for PBAPS, Units 2 and 3, has made changes to the facility over the life of the plant that may have invoked the final GDC. The extent to which the final GDC have been invoked can be found in specific sections of the UFSAR and in other plant-specific design and licensing basis documentation.

The NRC staff identified the following GDC as being applicable to the LAR:

- Draft GDC 1, "Quality Standards (Category A)," which requires, in part, that those systems and components that are essential to the prevention of accidents, which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, and erected to quality standards commensurate with the importance of the safety function to be performed.

- Draft GDC 4, "Sharing of Systems (Category A)," which requires that reactor facilities not share systems or components unless it is shown that safety is not impaired by the sharing.

- Draft GDC 8, "Overall Power Coefficient (Category B)," which requires that the reactor be designed so that the overall power coefficient in the power operating range shall not be positive.

- Draft GDC 9, "Reactor Coolant Pressure Boundary (Category A)," which requires that the reactor coolant pressure boundary (RCPB) be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

- Draft GDC 10, "Containment (Category A)," which requires that the containment structure be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features (ESFs) as may be necessary, to retain functional capability for as long as the situation requires.

- Draft GDC 12, "Instrumentation and Control Systems (Category B)," which requires that instrumentation and controls be provided as required to monitor and maintain variables within prescribed operating ranges.

- Draft GDC 14, "Core Protection Systems (Category B)," which requires that core protection systems, together with associated equipment, be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

- Draft GDC 15, "Engineered Safety Features Protection Systems (Category B)," which requires that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs.

- Draft GDC 19, "Protection Systems Reliability (Category B)," which requires that protection systems be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

- Draft GDC 20, "Protection Systems Redundancy and Independence (Category B)," which requires, in part, that redundancy and independence designed into protection systems be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function.

- Draft GDC 22, "Separation of Protection and Control Instrumentation Systems (Category B)," which requires that protection systems be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system, satisfying all requirements for the protection channels.

- Draft GDC 25, "Demonstration of Functional Operability of Protection Systems (Category B)," which requires that means be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

- Draft GDC 26, "Protection Systems Fail-Safe Design (Category B)," which requires that the protection system be designed to fail in a safe state or into a state established as tolerable

on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

- Draft GDC 27, "Redundancy of Reactivity Control (Category A)," which requires that at least two independent reactivity control systems, preferably of different principles, be provided.

- Draft GDC 28, "Reactivity Hot Shutdown Capability (Category A)," which requires that at least two of the reactivity control systems provided be independently capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

- Draft GDC 29, "Reactivity Shutdown Capability (Category A)," which requires, in part, that at least one of the reactivity control systems provided be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits.

- Draft GDC 30, Reactivity Holddown Capability (Category B)," which requires that at least one of the reactivity control systems provided be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

- Draft GDC 31, "Reactivity Control Systems Malfunction (Category B)," which requires that the reactivity control systems be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient that could result in exceeding acceptable fuel damage limits.

- Draft GDC 32, "Maximum Reactivity Worth of Control Rods (Category A)," which requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the RCPB or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

- Draft GDC 37, "Engineered Safety Features Basis for Design (Category A)," which requires, in part, that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems.

- Draft GDC 40, "Missile Protection (Category A)," which requires that protection for ESFs be provided against dynamic effects and missiles that might result from plant equipment failures.

- Draft GDC 42, "Engineered Safety Features Components Capability (Category A)," which requires that ESFs be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident (LOCA).

- Draft GDC 41, "Engineered Safety Features Performance Capability (Category A)," which requires, in part, that ESFs such as emergency core cooling and containment heat removal

systems provide the required safety function, assuming a failure of a single active component.

- Draft GDC 49, "Containment Design Basis (Category A)," which requires, in part, that the containment be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA.

- Draft GDC 51, "Reactor Coolant Pressure Boundary Outside Containment (Category A)," which requires, in part, that if part of the RCPB is outside the containment, appropriate features as necessary be provided to protect the health and safety of the public in case of an accidental rupture in that part.

- Draft GDC 57, "Provisions for Testing of Isolation Valves (Category A)," which requires, in part, that capability be provided for testing the operability of isolation valves to determine that valve leakage does not exceed acceptable limits.

- Final GDC 4, "Environmental and Dynamic Effects Design Bases," which requires, in part, that structures, systems, and components (SSCs) important to safety be protected against dynamic effects.

- Final GDC 10, "Reactor design," which requires, in part, that the reactor protection system (RPS) be designed to assure that specified, acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

- Final GDC 12, "Suppression of reactor power oscillations," which requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding specified, acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- Final GDC 19, "Control room," which requires, in part, 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.

- Final GDC 31, "Fracture prevention of reactor coolant pressure boundary," which requires, in part, that the RCPB be designed with sufficient margin to assure that when stressed under specified conditions, it will behave in a nonbrittle manner, and the probability of rapidly propagating fracture is minimized.

Technical Specification Requirements

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

As discussed in 10 CFR 50.36(c)(2), LCOs are the lowest functional capability or performance level of equipment required for safe operation of the facility. When LCOs are not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCOs can be met.

As discussed in 10 CFR 50.36(c)(3), SRs 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 LCOs will be met.

As discussed in 10 CFR 50.36(c)(5), administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

In general, there are two classes of changes to TSs: (1) changes needed to reflect contents of the design basis (TSs are derived from the design basis) and (2) voluntary changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TSs. The proposed amendments deal with the first class of change, namely, a change that is necessary to reflect the contents of the design basis.

Other Regulatory Requirements

The NRC staff identified the following regulatory requirements as being applicable to the LAR:

- 10 CFR Part 20, "Standards for Protection Against Radiation," which, in part, establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas and contains limits for occupational and public radiation doses.

- 10 CFR 50.44, "Combustible gas control for nuclear power reactors," which requires, in part, that plants be provided with the capability for 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, in part, establishes standards for the calculation of emergency core cooling system (ECCS) accident 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, in part that:

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

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

(3) Each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS. ATWS is defined as an AOO followed by the failure of the reactor trip portion of the protection system.

- 10 CFR 50.63, "Loss of all alternating current power," which requires, in part, that the plant withstand and recover from a station blackout (SBO) event of a specified duration.
- 10 CFR 50.67, "Accident source term," which, in part, sets limits for the radiological consequences of a postulated design-basis accident (DBA) using an alternative source term (AST). The NRC approved a full scope implementation of an AST methodology for PBAPS, Units 2 and 3, by License Amendment Nos. 269 and 273 on September 5, 2008 (ADAMS Accession No. ML082320406).
- 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," which, in part, sets numerical guides to meet the "as low as is reasonably achievable" (ALARA) criterion.
- 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," which, in part, establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA.

Guidance Documents

The guidance that the NRC staff considered in its review of this LAR included the following:

- NRC, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001, Revision 0, dated December 2003 (Reference 15).
- NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, dated September 30, 2007 (Reference 16).
- NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, dated November 2012 (Reference 17).
- NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980 (Reference 18).
- Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 4, dated March 2012 (Reference 19).
- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, dated May 2011 (Reference 20).
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 21).

- NRC Staff Requirements Memorandum for SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated July 21, 1993 (Reference 22).

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereinafter referred to as the SRP). Relevant sections of the SRP used in the review of this LAR are listed in Reference 23.

3.0 TECHNICAL EVALUATION

3.1 Overview of the SAR

The PBAPS MELLLA+ SAR, GEH report NEDC-33720P, Revision 0 (Attachment 4 to the application dated September 4, 2014), contains information divided into the following sections:

- SAR Section 1.0 – Introduction
- SAR Section 2.0 – Reactor Core and Fuel Performance
- SAR Section 3.0 – Reactor Coolant and Connected Systems
- SAR Section 4.0 – Engineered Safety Features
- SAR Section 5.0 – Instrumentation and Control
- SAR Section 6.0 – Electrical Power and Auxiliary Systems
- SAR Section 7.0 – Power Conversion Systems
- SAR Section 8.0 – Radwaste Systems and Radiation Sources
- SAR Section 9.0 – Reactor Safety Performance Evaluations
- SAR Section 10.0 – Other Evaluations
- SAR Section 11.0 – Licensing Evaluations
- SAR Section 12.0 – References

The SAR also contains the following Appendices:

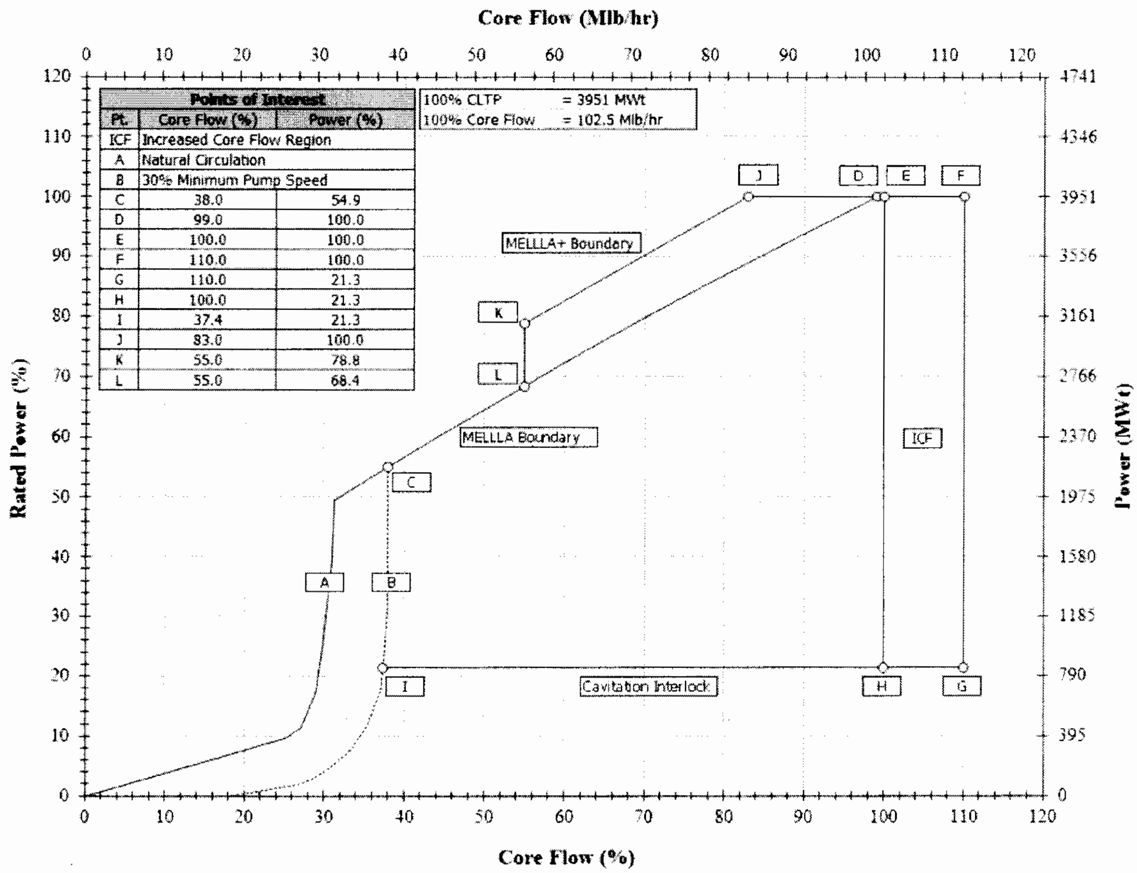
- Limitations from NRC SE for LTR NEDC-33173P-A (Methods LTR)
- Limitations from NRC SE for LTR NEDC-33006P-A (M+ LTR)
- Limitations from NRC SE for LTR NEDC-33075P-A (DSS-CD LTR)
- Limitations and Conditions Applicable to the use of TRACG / PANAC11 in ATWS Overpressure Analysis

As discussed in Section 1.0 of the SAR, the scope of the evaluations required to support operation in the MELLLA+ operating domain is contained in the M+ LTR. The SAR provides a systematic disposition of the M+ LTR topics applied to PBAPS, including performance of plant-specific assessments and confirmation of the applicability of those topics that were generically dispositioned in the M+ LTR.

Table 1-1 of the SAR lists the primary computer codes used to support the PBAPS plant-specific MELLLA+ evaluations. The evaluation of the use of these codes is contained in SE Section 3.4.4.

Figure 1-1 of the SAR (reproduced here as Figure 3.1-1) defines the proposed PBAPS MELLLA+ operating domain.

Figure 3.1-1 - MELLLA+ Operating Domain for PBAPS



The upper boundary of the MELLLA+ domain is defined by the following relation between the percent power (P) and the percent core flow (W_T):

[[ ]]

Section 1.2.4 of the SAR describes the allowed operational enhancements that are covered by the generically-approved M+ LTR. The following enhancements are allowed in the MELLLA+ operating domain and are proposed to be included for PBAPS:

- Increased core flow (ICF)
- Feedwater (FW) heater out-of-service (FWHOOS)
- Up to one safety relief valve (SRV) out-of-service (SRVOOS)
- Turbine bypass valves out-of-service
- End-of-cycle (EOC) recirculation pump trip out-of-service
- 24-month cycle

The following enhancements are not allowed in the MELLLA+ operating domain and are not proposed to be included for PBAPS:

- Final FW temperature reduction (FFWTR)
- Single loop operation (SLO)

3.2 <u>Generic MELLLA+ Dispositions</u>

3.2.1 Introduction

As discussed in SE Section 3.1 above, the SAR provides a systematic disposition of the M+ LTR topics applied to PBAPS, including performance of plant-specific assessments and confirmation of the applicability of those topics that were generically dispositioned in the M+ LTR.

As discussed in Section 1.1.1 of the SAR, generic assessments are those topics that can be disposed by either (1) providing or referencing a bounding analysis for the limiting conditions; (2) demonstrating that there is negligible effect due to MELLLA+ operation; (3) identifying the portions of the plant that are unaffected by the MELLLA+ power/flow map operating domain expansion; or (4) demonstrating that the sensitivity to MELLLA+ is small enough that the required plant-specific reload process is sufficient and appropriate for establishing the MELLLA+ licensing basis.

The NRC staff has briefly summarized the licensee's generic MELLLA+ dispositions for PBAPS in SE Sections 3.2.2 through 3.2.10. Section 3.2.11 of this SE provides the staff's conclusion regarding the licensee's generic MELLLA+ dispositions.

3.2.2 SAR Section 2.0 - Reactor Core and Fuel Performance

Section 2.0 of the SAR, "Reactor Core and Fuel Performance," indicates that each of the topics covered in this section was generically dispositioned in the M+ LTR. The SAR also indicates that the licensee's assessment determined that PBAPS met the M+ LTR disposition (i.e., generic assessments are applicable to PBAPS). As such, no plant-specific assessments were required for PBAPS.

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 2.0 of the SAR.

SAR Section 2.1.1 - Fuel Product Line

The current operating cores for both units contain only GNF2 fuel. At the time of implementation of the MELLLA+ amendment, the PBAPS, Units 2 and 3, cores will consist only of GNF2 fuel, which has been analyzed for use under MELLLA+ conditions. As discussed in SAR Section 2.1.1, the Supplemental Reload Licensing Report (SRLR) will confirm that there are no new fuel products as a result of MELLLA+ and will validate the conclusion that no additional fuel and core design evaluation is required for PBAPS.

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

SAR Section 2.1.2 - Core Design and Fuel Thermal Monitoring Threshold

As discussed in SAR Section 2.1.2, there is no change to the average power density as a result of MELLLA+ operating domain expansion. Because the maximum licensed power level and fuel product line design do not change as a result of MELLLA+, there is no increase in the average bundle power or in the maximum allowable peak bundle power. Because there is no change in average power density, there is no change required to the fuel thermal monitoring threshold.

As shown in SAR Table 2-2, the local power range monitor (LPRM) bypass void level is predicted to be zero and is thereby below the [[ ]] design requirement.

Detailed operating conditions (e.g., peak exposure, powers, void, and linear heat generation rate (LHGR)) have been provided as required by the M+ LTR.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because consistent with the generic disposition, the maximum licensed power level and fuel product line design do not change as a result of MELLLA+. In addition, the licensee confirmed that the predicted bypass void fraction satisfies the design requirement.

SAR Section 2.2.1 - Safety Limit Minimum Critical Power Ratio

The safety limit minimum critical power ratio (SLMCPR) is calculated based on the actual core loading pattern for each reload core in accordance with the methods defined in the GNF Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (Reference 24). This topical report (referred to as the GESTAR II process) is referenced in PBAPS TS 5.6.5, "Core Operating Limits Report (COLR)," as a methodology approved by the NRC as being applicable for determination of the PBAPS core operating limits.

As discussed in SAR Section 2.2.1, as required by the M+ LTR Limitation and Condition 12.6, the SLMCPR is calculated at Statepoints E, J, K, and F (shown in Figure 3.1-1 of this SE). In addition, uncertainties for SLO are applied to Statepoints J and K. The calculated values will be documented in the SRLR.

The licensee stated in SAR Section 2.2.1 that there are statepoints where the power-to-flow ratio exceeds 42 megawatts thermal per million pounds mass per hour (MWt/Mlbm/hr). As required by the Methods LTR Limitation and Condition 9.5, for MELLLA+ operation with a power-to-flow ratio greater than 42 MWt/Mlbm/hr, a +0.02 SLMCPR adder will be added to the cycle-specific SLMCPR, determined based on M+ LTR Limitation and Condition 12.6. As such, for PBAPS, the cycle-specific SLMCPR analysis will incorporate a +0.02 SLMCPR adder for MELLLA+ operation. The calculated values will be documented in the SRLR.

In accordance with the M+ LTR, using the methods in the GESTAR II process, the cycle-specific SLMCPR is determined, and a TS change will be requested, by the licensee if the current TS value is not bounding. Subsequent to submittal of the MELLLA+ application, by letters dated December 14, 2014 (Reference 25) and April 30, 2015 (Reference 26), the license requested

changes to the SLMCPR TSs for PBAPS, Units 2 and 3, respectively. Both LARs were submitted to support operation in the MELLLA+ operating domain during Cycle 21[1] operation for each of the units. The proposed amendments would change the SLMCPR values as shown in the following table:

Table 3.2.2-1 - Current and Proposed SLMCPR Values

Parameter	Current SLMCPR Value	Proposed SLMCPR Value
Unit 2 Two Recirculation Loop Operation	≥ 1.10	≥ 1.15
Unit 2 Single Recirculation Loop Operation	≥ 1.14	≥ 1.15
Unit 3 Two Recirculation Loop Operation	≥ 1.09	≥ 1.15
Unit 3 Single Recirculation Loop Operation	≥ 1.12	≥ 1.15

The proposed SLMCPR amendments would be implemented at the same time as the proposed MELLLA+ amendment.

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

SAR Section 2.2.2 - Operating Limit Minimum Critical Power Ratio

As discussed in SAR Section 2.2.2, the operating limit minimum critical power ratio (OLMCPR) is determined on a cycle-specific basis from the results of the reload transient analysis, in accordance with the GESTAR II process. The cycle-specific analyses are documented in the SRLR and included in the COLR. The MELLLA+ operating conditions do not change the methods used to determine this limit.

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

SAR Section 2.2.3 - Maximum Average Planar Linear Heat Generation Rate Limits

As discussed in SAR Section 2.2.3, the maximum average planar linear heat generation rate (MAPLHGR) operating limit is determined [[
]] The MELLLA+ operating conditions do not change the methods used to determine this limit.

[1] PBAPS, Unit 2, entered Cycle 21 following completion of the fall 2014 refueling outage. PBAPS, Unit 3, entered Cycle 21 following completion of the fall 2015 refueling outage.

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

SAR Section 2.2.4 - Linear Heat Generation Rate

As discussed in SAR Section 2.2.4, the linear heat generation rate (LHGR) limit is determined by the fuel rod thermal-mechanical (T-M) design and is not affected by MELLLA+ operating conditions. There are no changes to the PBAPS fuel or fuel design limits as a result of MELLLA+. [[
]] The MELLLA+ operating conditions do not change the methods used to determine this limit.

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

SAR Section 2.2.5 - Power-to-Flow Ratio

As discussed in SAR Section 2.2.5, the Methods LTR Limitation and Condition 9.3 requires that plant-specific EPU and expanded operating domain applications confirm that the core thermal power to core flow (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 licensee's LAR must include a power distribution assessment to establish that axial and nodal power distribution uncertainties determined via neutronic methods have not increased.

As shown in SAR Table 2-3, and as discussed in SAR Section 1.2.1, each point in the power-to-flow map is in compliance with the Methods LTR Limitation and Condition 9.3 threshold of 50 MWt/Mlbm/hr, with the exception of the Statepoint K (55 percent of rated core flow (RCF) and 78.8 percent of CLTP). At Statepoint K, the power-to-flow ratio is 55.23 MWt/Mlbm/hr. Section 2.2.5 of the SAR addressed this issue as follows:

> [[
>
>]]

The NRC staff review regarding the acceptability of exceeding the 50 MWt/Mlbm/hr threshold is evaluated in SE Section 3.4.4.

SAR Section 2.3.1 - Hot Excess Reactivity

As discussed in SAR Section 2.3.1, operation in the MELLLA+ operating domain may change the hot excess reactivity during the cycle. [[

]] The MELLLA+ operating conditions do not change the methods used to evaluate hot excess reactivity.

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

]]

SAR Section 2.3.2 - Strong Rod Out Shutdown Margin

As discussed in SAR Section 2.3.2, the PBAPS current design and TS cold shutdown margin limits are unchanged by MELLLA+. The MELLLA+ operating conditions do not change the PBAPS methods used to evaluate that strong rod out (SRO) shutdown margin meets the current PBAPS design and TS cold shutdown limits. [[

]]

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

]]

SAR Section 2.3.3 - Standby Liquid Control System Shutdown Margin

As discussed in SAR Section 2.3.3, the PBAPS current design and standby liquid control system (SLC) system TS requirements are unchanged by MELLLA+. The MELLLA+ operating conditions do not change the PBAPS methods used to evaluate that SLC system shutdown margin meets the current PBAPS design and SLC system TS requirements. [[

]]

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

]]

SAR Section 2.4 - Stability

The MELLLA+ core operating domain expansion does not require major plant system modifications. However, areas on the operating power/core flow map exist in which there is a potential for core instability with neutron flux oscillations, which have the possibility to grow in amplitude. These potential uncontrolled oscillations could cause safety limits such as SLMCPR to be exceeded. Instrumentation and control (I&C) systems are needed to identify any potential instability, indicate the condition, and automatically initiate corrective action.

As discussed in PBAPS UFSAR Section 7.5.7, the average power range monitor (APRM) system provides a continuous indication of average reactor power from a few percent to 125 percent of reactor power. The APRM system has four channels, each of which uses input signals from 43 local power range monitor (LPRM) detectors. Each APRM channel includes an oscillation power range monitor (OPRM) upscale function that monitors small groups of LPRM signals to detect thermal-hydraulic instabilities.

Each APRM channel provides trip signals to the reactor protection system (RPS). Any two unbypassed APRM channels, via the APRM two-out-of-four voter channels, can initiate an RPS trip in both RPS trip systems.

The OPRM software currently has three algorithms for detecting thermal-hydraulic instability related to neutron flux oscillations: the period based detection algorithm (PBDA), the amplitude based algorithm (ABA), and the growth rate algorithm (GRA). An OPRM upscale trip output is generated from an APRM channel when the PBDA in that channel detects oscillatory changes in the neutron flux. The PBDA is currently credited in the PBAPS licensing basis for the OPRM. An OPRM upscale trip is also generated from an APRM channel if either the ABA or GRA detect growing oscillatory changes in the neutron flux. However, the ABA and GRA are not credited in the licensing basis for the OPRM. They are provided for defense-in-depth only.

As discussed in the DSS-CD LTR, following a March 1988 instability event at a BWR, the BWR Owners' Group (BWROG) initiated a task to investigate actions that industry should take to resolve the stability issue as an operational concern. Through analysis, the BWROG found that the existing plant protection system, which was based on a scram on high APRM signal, may not provide enough protection against out-of-phase modes of instability. Thus, the BWROG decided that a new automatic instability suppression function was required as a long-term solution and that this function should have a rapid and automatic response, which does not rely on operator action. The BWROG submitted, and the NRC staff approved, three different long-term stability options. PBAPS implemented the "Option III" long-term stability solution. Option III uses the PBDA, ABA, and GRA detection algorithms for the OPRM.

The proposed MELLLA+ for PBAPS includes implementation of the detect and suppress solution - confirmation density (DSS-CD) stability solution for the OPRM. In addition to the existing three algorithms in the Option III stability solution, the DSS-CD uses an enhanced detection algorithm called the confirmation density algorithm (CDA). DSS-CD can be implemented as a software change using the existing GEH nuclear measurement analysis and control (NUMAC) power range neutron monitoring (PRNM) hardware currently used for Option III.

As discussed in SAR Section 2.4, the DSS-CD stability solution is designed to provide an early trip signal upon instability inception prior to any significant oscillation amplitude growth and MCPR degradation for both core-wide and regional mode oscillations. The licensee stated that PBAPS will implement the DSS-CD solution consistent with the M+ LTR and the DSS-CD LTR. In accordance with DSS-CD LTR Limitation and Condition 5.1, because PBAPS is implementing DSS-CD using the NRC-approved GEH Option III platform, a plant-specific review is not required. The specific topics pertaining to the DSS-CD stability solution are discussed below (SAR Sections 2.4.1, 2.4.1.1, 2.4.2, 2.4.3, and 2.4.4).

SAR Section 2.4.1 - DSS-CD Setpoints

As discussed in SAR Section 2.4.1, [[

]]

As a part of DSS-CD implementation, an applicability checklist, included in the DSS-CD LTR, is incorporated into the reload evaluation process and is documented in the SRLR. DSS-CD implementation also includes incorporation of appropriate [[ ]] analyses to be performed if a specific reload analysis [[ ]] As stated in the SAR, [[ ]] no further review of MELLLA+ is necessary to evaluate the adequacy of the DSS-CD setpoints.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the plant-specific reload process, [[ ]] is sufficient for establishing the DSS-CD setpoints.

SAR Section 2.4.1.1 - DSS-CD Diversity

The M+ LTR does not specifically address DSS-CD diversity. However, in SAR Section 2.4.1.1, the licensee has addressed this topic as part of the generic assessment included for the stability topic.

SAR Section 2.4.1.1 states that the licensee performed an evaluation for PBAPS regarding a potential common cause failure (CCF) of the PRNM system, which would disable the OPRM with the DSS-CD. The purpose of this evaluation was to determine if sufficient diversity exists so that the plant has the ability to cope with any CCF in the PRNM system.

Guidance for evaluating diversity and defense-in-depth (D3) of digital I&C systems is contained in NRC NUREG-0800, Branch Technical Position (BTP) 7-19 (Reference 23(l)). Section 1.4 provides the following four points as guidance in evaluating D3:

Point 1
The applicant shall assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common-mode failures have adequately been addressed.

Point 2
In performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report using best-estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.

Point 3
If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same

common-mode failure, shall be required to perform either the same function or a different function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.

Point 4
A set of displays and controls located in the main control room shall be provided for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in Items 1 and 3 above.

The NRC staff review of DSS-CD diversity, using the guidance in BTP 7-19, follows.

Point 1 in BTP 7-19 calls for assessment of the defense-in-depth and diversity of the proposed I&C system. The DSS-CD for PBAPS would retain the existing Option III algorithms (with generic setpoints). Thus, three additional algorithms are included for detecting thermal-hydraulic instability related neutron flux oscillations. All four algorithms are implemented in the OPRM upscale trip function, but the safety analysis, after implementation of MELLLA+, would only take credit for the CDA. The OPRM upscale trip function OPERABILITY would be based only on the CDA. The remaining three algorithms would provide defense-in-depth and additional protection against unanticipated oscillations. An OPRM upscale trip would also be issued from the channel if any of the defense-in-depth algorithms (PBDA, ABA, GRA) exceed their trip condition for one or more cells in that channel. Two-out-of-four channel voter logic is required for a reactor trip.

Failure of the NUMAC OPRM or APRM could disable the automatic safety trip function performed by the DSS-CD algorithms. However, the PBAPS NUMAC system provides an alternate means of accomplishing this stability safety protection function, via the automatic backup stability protection (ABSP), in the event that the primary means of stability protection (DSS-CD) becomes inoperable.

Use of common software for both primary (DSS-CD) and backup (ABSP) stability protection could lead to a postulated condition where both of these automatic functions would become disabled. If a latent software defect is assumed to exist, and under certain conditions (a trigger) results in this latent software defect preventing redundant, but otherwise independent protection functions from accomplishing the stability protection, this would be a CCF. The postulated CCF, assumed to result in comprehensive loss of PRNM system functionality, would also disable the OPRM system (i.e., CDA for DSS-CD and PBDA for Option III). The loss of PRNM system functionality would also disable the ABSP function of DSS-CD because the APRM system would no longer be available.

Point 3 in BTP 7-19 states, in part, that if a postulated CCF could disable a safety function, then a diverse means is to be provided, which is unlikely to be subject to the same CCF and shall be required to perform either the same function or a different function.

Point 4 in BTP 7-19 states, in part, that controls shall be independent and diverse from the safety computer system identified in Points 1 and 3. If the OPRM system is inoperable and the ABSP function performed by the APRM either cannot be implemented or is inoperable, manual backup stability protection (BSP) becomes the licensed stability solution (i.e., the diverse

means). The PBAPS power-recirculation flow graph contains regions of operation that are defined by a BSP Scram Region boundary. When plant conditions exceed this BSP Scram Region boundary, administrative actions require the nuclear reactor operators to initiate a manual reactor scram. This manual scram is accomplished by means independent of the software and systems associated with the CCF of the stability protection systems. This is essentially the same backup approach used in Option III for the PBDA algorithm. In the Option III solution, there is only one BSP option, which is provided by the manual BSP regions and associated reactor operator actions.

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

Point 2 in BTP 7-19 states, in part, that adequate diversity within the design be demonstrated for events that create the conditions for potential instability concurrent with a postulated CCF failure of the stability protection system. The NRC staff observed in a BWR that if the reactor is at RTP and there is a sudden loss of a significant amount of core flow, then the reactor could be placed in the region of the power/flow map subject to potential instability. Since the MELLLA+ operating domain is not analyzed for single recirculation loop operation, the proposed amendment would revise TS 3.4.1, "Recirculation Loops Operating," to prohibit single recirculation pump operation in the MELLLA+ domain and would require immediate action to exit the MELLLA+ domain if the plant was operating with a single recirculation loop. In addition, PBAPS emergency operating procedures (EOPs) require immediate action to reduce reactor power in order to mitigate possible high growth rate power oscillations following unanticipated significant loss of core flow events.

As stated in SAR Section 2.4.1.1, [[

]]

Section 2.4.1.1 of the SAR explains that PBAPS EOPs require immediate action to reduce reactor power in order to mitigate possible high growth-rate power oscillations following unanticipated core flow reduction events, such as [[ ]] event. Besides the indications that a 2RPT occurred, the operators would know the statepoint because the status of recirculation pumps is provided independent of the PRNM system; flow information is available from the recirculation flow system, and power level information is available from either the electrical power output or a core thermal power calculation. Furthermore, the reactor recirculation flow system, rod position information system, reactor manual control system, and manual scram are unaffected by the CCF. Thus, the plant is able to cope with the CCF because the operators can determine that defensive steps are necessary and execute those steps via immediate actions [[

]] Because the SLMCPR is not exceeded throughout this event, the acceptance criteria provided in BTP 7-19 are met.

Section 2.4.1.1 of the SAR addresses [[

]] Because the SLMCPR is not exceeded throughout this event, the acceptance criteria provided in BTP 7-19 are met.

In its previous evaluation of BSP protection, for the DSS-CD LTR, the NRC staff concluded that the proposed BSP methodology is an acceptable solution, because it provides sufficient protection against plant SLMCPR violations commensurate with the probability of an instability event in the short period of time they are active. With respect to PBAPS, the NRC staff further concludes that the manual control measures needed to support BSP protection are sufficiently diverse from the digital PRNMS NUMAC systems and, therefore, provide an acceptable means of diverse protection for the DSS-CD safety function in accordance with the guidance of BTP 7-19.

SAR Section 2.4.2 - Armed Region

As discussed in the M+ LTR, the OPRM trip-enabled region is termed the Armed Region. In the DSS-CD LTR, the Armed Region boundaries are specified to conservatively envelope power and flow conditions potentially susceptible to power oscillation. The trip function is enabled below a specified core flow and above a specified core power.

As discussed in SAR Section 2.4.2, per the DSS-CD LTR and the M+ LTR, the OPRM Armed Region is generically defined as the region on the power/flow map at the MCPR monitoring threshold of 25 percent OLTP and rated recirculation drive flow ≤ 75 percent. For a power-uprated plant, the MCPR monitoring threshold may be scaled to a lower percent value. For PBAPS, the MCPR monitoring threshold is 23.0 percent of EPU. As a result, the OPRM Armed Region for PBAPS is defined as the region on the power/flow map with power ≥ 23.0 percent of EPU and rated recirculation drive flow ≤ 75 percent.

The DSS-CD LTR generically specifies the Armed Region for MELLLA+ operation below 75 percent rated core flow and above 25 percent OLTP. For a power uprated plant (such as PBAPS), the setpoint is scaled to maintain the same power level in MWt.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because PBAPS is implementing the DSS-CD solution consistent with the DSS-CD LTR, and the OPRM Armed Region boundaries were generically approved as part of the DSS-CD LTR.

SAR Section 2.4.3 - Backup Stability Protection

As discussed in SAR Section 2.4.3, the DSS-CD LTR defines the BSP, along with a generic process for confirming the BSP requirements are met in each reload analysis. Implementation of DSS-CD in accordance with the DSS-CD LTR requires that PBAPS confirm that the BSP approach is adequate as part of the reload process.

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

SAR Section 2.5.1 - Control Rod Scram

As discussed in SAR Section 2.5.1, the generic disposition of the control rod scram topic in the M+ LTR describes that for BWR/3, BWR/4, and BWR/5 plants, 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. PBAPS is of the BWR/4 design. The PBAPS reactor dome pressure of 1,035 pounds per square inch gauge (psig) does not change as a result of MELLLA+ operation.

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

SAR Section 2.5.2 - Control Rod Drive Positioning and Cooling

As discussed in SAR Section 2.5.2, consistent with the generic disposition of the control rod drive (CRD) positioning and cooling topic in the M+ LTR, for PBAPS, [[ ]]]] In addition, the licensee stated that reactor coolant temperature does not change for MELLLA+ operation.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because there is negligible effect due to MELLLA+ operation on the CRD positioning and cooling functions.

SAR Section 2.5.3 - Control Rod Drive Integrity

As discussed in SAR Section 2.5.3, the generic disposition of the CRD integrity topic in the M+ LTR describes 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. The licensee stated that, for PBAPS, [[ ]]

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

3.2.3 SAR Section 3.0 - Reactor Coolant and Connected Systems

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 3.0 of the SAR.

SAR Section 3.1.1 - Flow-Induced Vibration

As discussed in 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 PBAPS. 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 PBAPS because FIV of piping and the SRVs is unaffected by operation in the MELLLA+ operating domain.

SAR Section 3.2.2 - Reactor Vessel Structural Evaluation

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

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

SAR Section 3.3.1.1 - Fuel Assembly Lift Forces

As discussed in SAR 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 PBAPS. Specifically, there are no significant changes in the core exit steam flow, reactor operating pressure, FW flow rates, or steam flow rates for MELLLA+ operation. The only variable affecting forces on the fuel assemblies in the MELLLA+ operating domain for normal, upset, emergency, and faulted conditions is the core flow. Maximum core flow is reduced in the MELLLA+ operating domain. As such, the lift forces for normal, upset, emergency, and faulted conditions in the MELLLA+ operating domain [[ ]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] with respect to fuel assembly lift forces.

SAR Section 3.3.1.2 - Reactor Internal Pressure Differences for Normal, Upset, Emergency and Faulted Conditions

As discussed in SAR Section 3.3.1.2, the licensee confirmed that the generic disposition in the M+ LTR for the reactor internal pressure differences (RIPDs) topic is applicable to PBAPS. Specifically, there are no significant changes in the core exit steam flow, reactor operating pressure, FW flow rates, or steam flow rates for MELLLA+ operation. The only variable affecting RIPDs in the MELLLA+ operating domain for normal, upset, emergency, and faulted conditions is the core flow. Maximum core flow is reduced in the MELLLA+ operating domain. As such, the RIPDs for normal, upset, emergency, and faulted conditions in the MELLLA+ operating domain [[ ]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] with respect to RIPDs.

SAR Section 3.4.1 - Flow-Induced Vibration Influence on Piping

As discussed in SAR Section 3.4.1, the licensee confirmed that the generic disposition in the M+ LTR for the FIV influence on piping topic is applicable to PBAPS. 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, [[ ]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] with respect to FIV influence on piping.

SAR Section 3.4.2 - Flow-Induced Vibration Influence on Reactor Internals

As discussed in SAR 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 PBAPS. Specifically, the MELLLA+ operating domain results in decreased core and recirculation flow and no increase in MS or FW flow rates. As such, [[ ]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] 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.

SAR Section 3.5.1.1 - Main Steam and Feedwater Piping Inside Containment

As discussed in SAR 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 PBAPS. Specifically, MS and FW system temperatures, flows, and pressures at MELLLA+ operating conditions are bounded by the current plant operation temperatures, flows, and pressures. As such, the parameters are within the values used in the design of the piping and supports for worst case conditions.

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

SAR Section 3.5.1.2 - Reactor Recirculation and Control Rod Drive Systems

As discussed in SAR 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 PBAPS. Specifically, the reactor recirculation and CRD system temperatures, flows, and pressures are bounded by the current plant operation temperatures, flows, and pressures. As such, the

parameters are within the values used in the design of the piping and supports for worst case conditions.

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

SAR Section 3.5.1.3 - Other Reactor Coolant Pressure Boundary Piping Systems

As discussed in SAR 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 other reactor coolant pressure boundary (RCPB) piping systems topic is applicable to PBAPS. Specifically, the temperatures, flows, and pressures for these systems (CS, RHR/LPCI, SLCS, RPV head vent line, SRV discharge lines, 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 PBAPS because operation in the MELLLA+ domain is bounded by current plant operation with respect to the other RCPB piping systems.

SAR Section 3.5.2.1 - Main Steam and Feedwater Outside Containment

As discussed in SAR 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 PBAPS. Specifically, MS and FW system temperatures, flows, and pressures at MELLLA+ operating conditions are bounded by the current plant operation temperatures, flows, and pressures. As such, the parameters are within the values used in the design of the piping and supports for worst case conditions. 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 PBAPS because operation in the MELLLA+ domain is bounded by current plant operation with respect to MS and FW piping outside containment.

SAR Section 3.5.2.2.1 - Other Balance of Plant Piping Systems - RCIC, HPCI, CS, and RHR

As discussed in SAR 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 (reactor core isolation cooling (RCIC), high pressure coolant injection (HPCI), core spray (CS), and residual heat removal (RHR) systems) topic is applicable to PBAPS. 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 PBAPS 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 PBAPS because operation in the MELLLA+ domain is bounded by current plant operation with respect to these BOP piping systems.

SAR Section 3.5.2.2.2 - Other BOP Piping Systems - Offgas System and Neutron Monitoring System

As discussed in SAR 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 PBAPS. Specifically, there is no change in the PBAPS reactor operating pressure or power level at MELLLA+ operating conditions.

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

SAR Section 3.6.1 - Reactor Recirculation System Evaluation

As discussed in SAR 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 PBAPS. Specifically, for PBAPS, 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 528.4 °F; in the MELLLA+ operating domain, it is 523.3 °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 amendment, TS 3.4.1, "Recirculation Loops Operating," would be revised to prohibit single loop operation (SLO) in the MELLLA+ domain.

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

SAR Section 3.6.2 - Net Positive Suction Head

As discussed in SAR Section 3.6.2, the licensee confirmed that the generic disposition in the M+ LTR for the net positive suction head (NPSH) topic for the RRS is applicable to PBAPS. Specifically, for PBAPS, [[

]]

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

SAR Section 3.6.3 - Single Loop Operation

As discussed in SAR Section 3.6.3, consistent with the generic disposition in the M+ LTR, for PBAPS, SLO for the RRS will not be allowed in the MELLLA+ operating domain. Specifically, the licensee has proposed a change to TS 3.4.1, "Recirculation Loops Operating," which would prohibit SLO in the MELLLA+ domain. The proposed TS change is evaluated in SE Section 4.2.

SAR Section 3.7 - Main Steam Line Flow Restrictors

As discussed in SAR Section 3.7, the licensee confirmed that the generic disposition in the M+ LTR for the MSL flow restrictors topic is applicable to PBAPS. Specifically, there is no increase in PBAPS MS flow as a result of the MELLLA+ operating domain expansion.

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

SAR Section 3.8 - Main Steam Isolation Valves

As discussed in SAR Section 3.8, the licensee confirmed that the generic disposition in the M+ LTR for the main steam isolation valves (MSIVs) topic is applicable to PBAPS. Specifically, there is no increase in PBAPS MS pressure, flow, or pressure drop as a result of the MELLLA+ operating domain expansion.

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

SAR Section 3.9 - Reactor Core Isolation Cooling System

As discussed in SAR Sections 3.9.1, 3.9.2, and 3.9.3, the licensee confirmed that the generic disposition in the M+ LTR for the reactor core isolation cooling (RCIC) topic with respect to system hardware, system initiation, and NPSH is applicable to PBAPS. Specifically:

- There are no changes to RCIC system hardware as a result of the MELLLA+ operating domain expansion.
- With respect to system initiation, there are no changes to the normal reactor operating pressure, decay heat, or SRV setpoints as a result of the MELLLA+ operating domain expansion.
- With respect to NPSH, there are no physical changes to the RCIC pump suction configuration as a result of the MELLLA+ operating domain expansion. There are also no changes to the RCIC flow rate or minimum atmospheric pressure in the suppression chamber and the condensate storage tank.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the RCIC system is unaffected by operation in the MELLLA+ operating domain.

SAR Section 3.10.2 - Suppression Pool and Containment Spray Cooling Modes

As discussed in SAR Section 3.10.2, the licensee confirmed that the generic disposition in the M+ LTR for the RHR suppression pool and containment spray cooling modes topic is applicable to PBAPS. Specifically, these modes of RHR are unaffected by MELLLA+ operation because [[ ]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the RHR suppression pool and containment spray cooling modes are unaffected by operation in the MELLLA+ operating domain.

SAR Section 3.10.3 - Shutdown Cooling Mode

As discussed in SAR Section 3.10.3, the licensee confirmed that the generic disposition in the M+ LTR for the RHR shutdown cooling mode topic is applicable to PBAPS. Specifically, this mode of RHR is unaffected by MELLLA+ operation because [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the RHR shutdown cooling mode is unaffected by operation in the MELLLA+ operating domain.

SAR Section 3.11.1 - Reactor Water Cleanup System Performance

As discussed in SAR Section 3.11.1, the licensee confirmed that the generic disposition in the M+ LTR for the reactor water cleanup (RWCU) system performance topic is applicable to PBAPS. Specifically, because there is no change to the pressure or fluid thermal conditions experienced by the RWCU system, and because there is no increase in the quantity of fission products, corrosion products, and other soluble and insoluble impurities in the reactor water, the implementation of MELLLA+ has no effect on the RWCU system.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the RWCU system is unaffected by operation in the MELLLA+ operating domain.

SAR Section 3.11.2 - Reactor Water Cleanup System Containment Isolation

As discussed in SAR Section 3.11.2, the licensee confirmed that the generic disposition in the M+ LTR for the RWCU system containment isolation topic is applicable to PBAPS. Specifically, as shown in SAR Table 1-2, there are no significant changes in FW line temperature, pressure, or flow rate.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain [[
]]

3.2.4 SAR Section 4.0 - Engineered Safety Features

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 4.0 of the SAR.

SAR Section 4.1.1.1 - Long-Term Suppression Pool Cooling Temperature Response

As discussed in SAR Section 4.1.1.1, the licensee confirmed that the generic disposition in the M+ LTR for the long-term suppression pool cooling temperature response topic is applicable to PBAPS. Specifically, for PBAPS, the sensible and decay heat do not change as a result of

MELLLA+ operation. Therefore, [[
]]

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

]]

SAR Section 4.1.2.3 - Safety Relief Valve Piping - Containment Dynamic Loads

As discussed in SAR Section 4.1.2.3, the licensee confirmed that the generic disposition in the M+ LTR for the safety relief valve (SRV) piping containment dynamic loads topic is applicable to PBAPS. Specifically, for PBAPS, the sensible and decay heat do not change as a result of MELLLA+ operation. In addition, the SRV setpoints do not change.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the SRV piping containment dynamic loads are unaffected by operation in the MELLLA+ operating domain.

SAR Section 4.1.2.4 - SRV Containment Dynamic Loads

As discussed in SAR Section 4.1.2.4, the licensee confirmed that the generic disposition in the M+ LTR for the SRV containment dynamic loads topic is applicable to PBAPS. Specifically, [[

]]

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

]]

SAR Section 4.1.3 - Containment Isolation

As discussed in SAR Section 4.1.3, the licensee confirmed that the generic disposition in the M+ LTR for the containment isolation topic is applicable to PBAPS. Specifically, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] with respect to containment isolation.

SAR Section 4.1.4 - Generic Letter 89-10

As discussed in SAR Section 4.1.4, the licensee confirmed that the generic disposition in the M+ LTR for Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" (Reference 28), is applicable to PBAPS. Specifically, [[

]] The licensee also confirmed that other parameters with the potential to affect the

capability of safety-related motor-operated valves (MOVs), such as the ambient temperature profile, are unchanged.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] with respect to the issues associated with GL 89-10.

SAR Section 4.1.5 - Generic Letter 89-16

As discussed in SAR Section 4.1.5, the licensee confirmed that the generic disposition in the M+ LTR for GL 89-16, "Installation of a Hardened Wetwell Vent" (Reference 29), is applicable to PBAPS. In response to GL 89-16, some plants, such as PBAPS, installed a hardened wetwell vent system to mitigate conditions in which the containment integrity is threatened by an overpressure condition related to the long-term loss of decay heat. One of the design requirements of the hardened vent system is the ability to exhaust energy equivalent to 1 percent of the current licensed thermal power. The licensee stated that [[

]]

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

]]

SAR Section 4.1.6 - Generic Letter 95-07

As discussed in SAR Section 4.1.6, the licensee confirmed that the generic disposition in the M+ LTR for GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves" (Reference 30), is applicable to PBAPS. Specifically, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] with respect to the issues associated with GL 95-07.

SAR Section 4.1.7 - Generic Letter 96-06

As discussed in SAR Section 4.1.7, the licensee confirmed that the generic disposition in the M+ LTR for GL 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions" (Reference 31), is applicable to PBAPS. Specifically, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] with respect to the issues associated with GL 96-06.

SAR Section 4.2.1 - High Pressure Coolant Injection

As discussed in SAR Section 4.2.1, the licensee confirmed that the generic disposition in the M+ LTR for the high pressure coolant injection (HPCI) system topic is applicable to PBAPS. Specifically, there is no change to the reactor pressure as a result of MELLLA+ operation. In addition, sensible and decay heat and the SRV setpoints do not change.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the HPCI system is unaffected by operation in the MELLLA+ operating domain.

SAR Section 4.2.3 - Core Spray

As discussed in SAR Section 4.2.3, the licensee confirmed that the generic disposition in the M+ LTR for the CS system topic is applicable to PBAPS. Specifically, there is no change to the reactor pressure as a result of operating in the MELLLA+ domain. In addition, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the CS system is unaffected by operation in the MELLLA+ operating domain.

SAR Section 4.2.4 - Low Pressure Coolant Injection

As discussed in SAR Section 4.2.4, the licensee confirmed that the generic disposition in the M+ LTR for the low pressure coolant injection (LPCI) topic is applicable to PBAPS. Specifically, there is no change to the reactor pressure as a result of operating in the MELLLA+ domain. In addition, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the LPCI system is unaffected by operation in the MELLLA+ operating domain.

SAR Section 4.2.5 - Automatic Depressurization System

As discussed in SAR Section 4.2.5, the licensee confirmed that the generic disposition in the M+ LTR for the automatic depressurization system (ADS) topic is applicable to PBAPS. Specifically, the MELLLA+ operating domain expansion [[

]]

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

SAR Section 4.2.6 - Emergency Core Cooling System Net Positive Suction Head

As discussed in SAR Section 4.2.6, the licensee confirmed that the generic disposition in the M+ LTR for the emergency core cooling system (ECCS) NPSH topic is applicable to PBAPS. Specifically, the MELLLA+ operating domain expansion does not result in an increase in the heat addition to the suppression pool following a LOCA (both large and small breaks), station

blackout (SBO), or Appendix R event. This disposition is also applicable to other PBAPS events, such as loss of RHR normal shutdown cooling function (NSDC), stuck open relief valve (SORV), and shutdown and cool down of the second (non-accident) PBAPS unit during a DBA-LOCA (on the other PBAPS unit), concurrent with the loss of off-site power (LOOP) and the loss of an emergency diesel generator (EDG). [[

]] There are no physical changes in the piping or system arrangement. There is no increase in the heat addition to the suppression pool following a LOCA, loss of RHR NSDC, SORV, shutdown and cool down of the second (non-accident) PBAPS unit during a DBA-LOCA (on the other PBAPS unit), concurrent with LOOP and loss of EDG, SBO, or Appendix R event. For PBAPS, the licensee stated that the most limiting case for ECCS NPSH had been confirmed to occur at the long-term suppression pool temperature, [[ ]]

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

SAR Sections 4.3.4 and 4.3.5 - Local Cladding Oxidation and Core Wide Metal Water Reaction

As discussed in SAR Sections 4.3.4 and 4.3.5, the licensee confirmed that the generic dispositions in the M+ LTR for the local cladding oxidation and core-wide metal water reaction topics are applicable to PBAPS. As discussed in the M+ LTR, the peak cladding temperature (PCT) change due to MELLLA+ will be calculated on a plant-specific basis for the limiting large break LOCA to demonstrate compliance with the 2200 °F acceptance criterion of 10 CFR 50.46. [[

]] As discussed in Section 3.3.4 of this SE, the licensee's analysis for large and small break LOCAs, under MELLLA+ conditions, determined that the PCT would remain below the 2200 °F acceptance criterion.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because there is a negligible effect on local cladding oxidation and core-wide metal water reaction due to MELLLA+ operation

SAR Sections 4.3.6 and 4.3.7 - Coolable Geometry and Long-Term Cooling

As discussed in SAR Sections 4.3.6 and 4.3.7, the licensee confirmed that the generic disposition in the M+ LTR for the coolable geometry and long-term cooling topics are applicable to PBAPS. Specifically, the M+ LTR concludes that [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because coolable geometry and long-term cooling are unaffected by operation in the MELLLA+ operating domain.

SAR Section 4.3.8 - Flow Mismatch Limits

As discussed in SAR Section 4.3.8, the licensee confirmed that the generic disposition in the M+ LTR for the flow mismatch limits topic is applicable to PBAPS. As discussed in the SAR, [[

]] Therefore, the current recirculation drive flow mismatch limits for PBAPS remain acceptable in the MELLLA+ region.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the effect due to recirculation drive flow mismatch is bounded by the licensee's analysis supporting operation in the MELLLA+ operating domain.

SAR Section 4.4 - Main Control Room Atmosphere Control System

As discussed in SAR Section 4.4, the licensee confirmed that the generic disposition in the M+ LTR for the main control room atmosphere control system topic is applicable to PBAPS. Specifically, there is no change in the PBAPS source term or release rates as a result of MELLLA+ operating domain expansion. [[
]]

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

]]

SAR Section 4.5.1 - Standby Gas Treatment System Flow Capacity

As discussed in SAR Section 4.5.1, the licensee confirmed that the generic disposition in the M+ LTR for the standby gas treatment system (SGTS) flow capacity topic is applicable to PBAPS. Specifically, the design flow capacity of the PBAPS SGTS was selected to maintain the secondary containment at the required negative pressure to minimize the potential for exfiltration of air from the Reactor Building. In addition, [[

]]

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

SAR Section 4.5.2 - Iodine Removal Capability

As discussed in SAR Section 4.5.2, the licensee confirmed that the generic disposition in the M+ LTR for the SGTS iodine removal capacity topic is applicable to PBAPS. Specifically, the core fission product inventory is not changed by the MELLLA+ operating domain expansion and coolant activity levels, which are defined by the TSs, do not change, and thus, no change occurs in the SGTS adsorber iodine loading, decay heat rates, or iodine removal efficiency.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the SGTS iodine removal capability is unaffected by operation in the MELLLA+ operating domain.

3.2.5 SAR Section 5.0 - Instrumentation and Control

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 5.0 of the SAR.

SAR Section 5.1.1 - Average Power Range, Intermediate Range, and Source Range Monitors

As discussed in SAR Section 5.1.1, the licensee confirmed that the generic disposition in the M+ LTR for the average power range monitors (APRMs), intermediate range monitors (IRMs), and source range monitors (SRMs) topic is applicable to PBAPS. Specifically, the APRM output signals are calibrated to read 100 percent at the CLTP. [[ ]] At PBAPS, the SRMs and IRMs were replaced by the wide range neutron monitoring (WRNM) system. The WRNMs are adjusted to ensure adequate overlap with the APRMs. [[ ]]

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

SAR Section 5.1.2 - Local Power Range Monitors

As discussed in SAR Section 5.1.2, the licensee confirmed that the generic disposition in the M+ LTR for the local power range monitors (LPRMs) topic is applicable to PBAPS. Specifically, there is no change in the neutron flux experienced by the LPRMs resulting from operating in the MELLLA+ domain. As such, [[ ]]

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

SAR Section 5.1.3 - Rod Block Monitors

As discussed in SAR Section 5.1.3, the licensee confirmed that the generic disposition in the M+ LTR for the rod block monitors (RBM) topic is applicable to PBAPS. Specifically, 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 PBAPS because [[ ]]

SAR Section 5.1.4 - Rod Worth Minimizer

As discussed in SAR Section 5.1.4, the licensee confirmed that the generic disposition in the M+ LTR for the rod worth minimizer (RWM) topic is applicable to PBAPS. Specifically, the

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

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

SAR Section 5.1.5 - Traversing Incore Probes

As discussed in SAR Section 5.1.5, the licensee confirmed that the generic disposition in the M+ LTR for the traversing incore probes (TIPs) topic is applicable to PBAPS. Specifically, 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 PBAPS because the TIPs are unaffected by operation in the MELLLA+ operating domain.

SAR Section 5.2 – Balance-of-Plant Monitoring and Control

As discussed in SAR Sections 5.2.1 through 5.2.6, the licensee confirmed that the generic disposition in the M+ LTR for the balance-of-plant (BOP) monitoring and control topic is applicable to PBAPS. Specifically, 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 PBAPS because BOP monitoring and control devices is unaffected by operation in the MELLLA+ operating domain.

SAR Section 5.3.2 - Rod Block Monitor

As discussed in SAR Section 5.3.2, the licensee confirmed that the generic disposition in the M+ LTR for the rod block monitor (RBM) topic is applicable to PBAPS. Specifically, [[ ]]

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

3.2.6 SAR Section 6.0 - Electrical Power and Auxiliary Systems

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 6.0 of the SAR.

SAR Section 6.1 - Alternating Current Power

As discussed in SAR Section 6.1, the licensee confirmed that the generic disposition in the M+ LTR for the alternating current (AC) power topic is applicable to PBAPS. Specifically, MELLLA+ operation does not change the PBAPS reactor thermal power or the electrical output from the station. In addition, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the AC power system [[ ]]

SAR Section 6.2 - Direct Current Power

As discussed in SAR Section 6.2, the licensee confirmed that the generic disposition in the M+ LTR for the direct current (DC) power topic is applicable to PBAPS. Specifically, MELLLA+ operation does not change system requirements for control or motive power loads. As such, [[ ]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the DC power system [[ ]]

SAR Section 6.3.1 - Fuel Pool Cooling

As discussed in SAR Section 6.3.1, the licensee confirmed that the generic disposition in the M+ LTR for the spent fuel pool (SFP) cooling topic is applicable to PBAPS. Specifically, reactor power does not increase as a result of MELLLA+ operation. [[ ]]

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

SAR Section 6.3.2 - Crud Activity and Corrosion Products

As discussed in SAR Section 6.3.2, the licensee confirmed that the generic disposition in the M+ LTR for the SFP crud activity and corrosion products topic is applicable to PBAPS. Specifically, [[ ]]

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

SAR Section 6.3.3 - Radiation Levels

As discussed in SAR Section 6.3.3, the licensee confirmed that the generic disposition in the M+ LTR for the SFP radiation levels topic is applicable to PBAPS. Specifically, [[ ]]

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

SAR Section 6.3.4 - Fuel Racks

As discussed in SAR Section 6.3.4, the licensee confirmed that the generic disposition in the M+ LTR for the fuel racks topic is applicable to PBAPS. Specifically, reactor power does not increase as a result of MELLLA+ operation. [[
]]

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

SAR Section 6.4 - Water Systems

As discussed in SAR Section 6.4, the licensee confirmed that the generic disposition in the M+ LTR for the water systems topic is applicable to PBAPS. Specifically, MELLLA+ operation does not affect the performance of the safety-related emergency service water (ESW) 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 disposition is applicable to PBAPS because the performance of water systems are unaffected by operation in the MELLLA+ operating domain.

SAR Section 6.5.1 - Shutdown Margin

As discussed in SAR Section 6.5.1, the licensee confirmed that the generic disposition in the M+ LTR for the SLC system shutdown margin topic is applicable to PBAPS. Specifically, SLC system shutdown margin for PBAPS is calculated as a part of the standard reload process.

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

SAR Section 6.6 - Heating, Ventilation, and Air Conditioning

As discussed in SAR Section 6.6, the licensee confirmed that the generic disposition in the M+ LTR for the heating, ventilation, and air conditioning (HVAC) topic is applicable to PBAPS. Specifically, for PBAPS HVAC systems, the process temperatures and heat loads from motors and cables are bounded by the CLTP process temperatures and heat loads; therefore, they are within the design of the HVAC equipment chosen for the worst case conditions.

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

SAR Section 6.7 - Fire Protection

As discussed in SAR Section 6.7, the licensee confirmed that the generic disposition in the M+ LTR for the fire protection topic is applicable to PBAPS. Specifically: [[

]]

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

SAR Section 6.8 - Other Systems Affected

As discussed in SAR Section 6.8, the licensee confirmed that the generic disposition in the M+ LTR for the other systems affected topic is applicable to PBAPS. Specifically, the licensee performed a review to assure that the SAR included all systems that may be affected by the implementation of MELLLA+. The licensee has confirmed that those systems that are significantly affected by operation in the MELLLA+ domain are addressed in the SAR.

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

3.2.7 SAR Section 7.0 - Power Conversion Systems

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 7.0 of the SAR.

SAR Section 7.1 - Turbine-Generator

As discussed in SAR Section 7.1, the licensee confirmed that the generic disposition in the M+ LTR for the turbine-generator topic is applicable to PBAPS. Specifically, there is no change in the PBAPS reactor power level, reactor operating pressure, MS flow rates, or electrical output of the generator as a result of MELLLA+ operation. Therefore, there is no change to the PBAPS missile avoidance and protection analysis.

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

SAR Section 7.2 - Condenser and Steam Jet Air Ejectors

As discussed in SAR Section 7.2, the licensee confirmed that the generic disposition in the M+ LTR for the condenser and steam jet air ejectors topic is applicable to PBAPS. Specifically, there is no change in the PBAPS 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 disposition is applicable to PBAPS because the condenser and steam jet air ejectors are unaffected by operation in the MELLLA+ operating domain.

SAR Section 7.3 - Turbine Steam Bypass

As discussed in SAR Section 7.3, the licensee confirmed that the generic disposition in the M+ LTR for the turbine steam bypass topic is applicable to PBAPS. Specifically, there is no change in the PBAPS 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 disposition is applicable to PBAPS because the turbine steam bypass system is unaffected by operation in the MELLLA+ operating domain.

SAR Section 7.4 - Feedwater and Condensate Systems

As discussed in SAR Section 7.4, the licensee confirmed that the generic disposition in the M+ LTR for the feedwater and condensate topic is applicable to PBAPS. Specifically, there is no change in the PBAPS FW pressure, temperature, and flow rates, and the performance requirements for the feedwater and condensate systems are not changed by MELLLA+.

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

3.2.8 SAR Section 8.0 - Radwaste Systems and Radiation Sources

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 8.0 of the SAR.

SAR Section 8.1.2 - Waste Volumes

As discussed in SAR Section 8.1.2, the licensee confirmed that the generic disposition in the M+ LTR for the waste volumes topic is applicable to PBAPS. Specifically, there is no change in the PBAPS reactor power level and no increases in the MS or FW flow rates as a result of MELLLA+ operation. The moisture content of the MS leaving the vessel may increase while operating near the minimum core flow (CF) in the MELLLA+ operating domain. The amount of liquid water carried in the steam is called moisture carry-over (MCO). Higher MCO will result in an increase in the soluble and non-soluble radioactive species in the reactor water being transported from the reactor vessel to the turbine and secondary side of the plant. However, the MCO values under MELLLA+ conditions are bounded by the pre-MELLLA+ conditions as discussed in SE Section 3.3.3 (under SAR Section 3.3.3). Due to the very small increase in reactor MCO reaching the condenser, the condensate full flow filtration filter backwash frequency and volume are not changed, and the disposal frequency of the condensate demineralizer resins is not changed. Additionally, because the reactor water cleanup (RWCU) system is not affected by operation in the MELLLA+ operating domain, the RWCU filter demineralizer backwash frequency is not changed. Therefore, the PBAPS waste volumes will not be affected by operation in the MELLLA+ operating domain.

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

SAR Section 8.2.1 - Off-Site Release Rate

As discussed in SAR Section 8.2.1, the licensee confirmed that the generic disposition in the M+ LTR for the off-site release rate topic is applicable to PBAPS. Specifically, the PBAPS radiological release rate is administratively controlled to remain within existing release rate limits. In addition, none of the applicable parameters (e.g., fuel cladding performance, main condenser air inleakage, charcoal adsorber inlet dew point, charcoal adsorber temperature) are affected by MELLLA+ operation.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the offsite release rate is unaffected by operation in the MELLLA+ operating domain.

SAR Section 8.2.2 - Recombiner Performance

As discussed in SAR Section 8.2.2, the licensee confirmed that the generic disposition in the M+ LTR for the recombiner performance topic is applicable to PBAPS. Specifically, [[

]] The PBAPS-specific value for radiolytic gas flow rate does not change as a result of operating in the MELLLA+ domain.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because recombiner performance is unaffected by operation in the MELLLA+ operating domain.

SAR Section 8.3 - Radiation Sources in the Reactor Core

As discussed in SAR Section 8.3, the licensee confirmed that the generic disposition in the M+ LTR for the radiation sources in the reactor core topic is applicable to PBAPS. Specifically, reactor power does not increase as a result of MELLLA+ operation. The PBAPS core average exposure for MELLLA+ is [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because radiation sources in the reactor core [[

]]

SAR Section 8.4.1 - Coolant Activation Products

As discussed in SAR Section 8.4.1, the licensee confirmed that the generic disposition in the M+ LTR for the coolant activation products topic is applicable to PBAPS. Specifically, reactor power does not increase and the steam flow rate does not change as a result of MELLLA+ operation. [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because coolant activation products [[ ]]

SAR Section 8.6.1 - Plant Gaseous Emissions

As discussed in SAR Section 8.6.1, the licensee confirmed that the generic disposition in the M+ LTR for the plant gaseous emission topic is applicable to PBAPS. Specifically, reactor power does not increase, and the steam flow rate does not change, as a result of MELLLA+ operation. [[ ]] The small increase in MCO, from periodically operating at or near the MELLLA+ minimum CF rate, results in a small increase in soluble radioactive iodine and particulates in airborne releases. However, these increases are within the current licensing basis.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because operation in the MELLLA+ domain is [[ ]] with respect to plant gaseous emissions.

SAR Section 8.6.2 - Gamma Shine from the Turbine

As discussed in SAR Section 8.6.2, the licensee confirmed that the generic disposition in the M+ LTR for the gamma shine from the turbine topic is applicable to PBAPS. Specifically, the PBAPS steam flow rate does not change as a result of MELLLA+ operation. In addition, the slight increase in moisture content in the reactor steam for MELLLA+ operation will not significantly affect the Nitrogen-16 (N-16) activity concentration (in units of microcuries per gram), because the total N-16 amount contained in the moisture is small compared to that contained in the dry steam.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because there is negligible effect by operation in the MELLLA+ operating domain with respect to gamma shine from the turbine.

3.2.9 SAR Section 9.0 - Reactor Safety Performance Evaluations

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 9.0 of the SAR.

SAR Section 9.1.3 - Non-Limiting Events

As discussed in SAR Section 9.1.3, the licensee confirmed that the generic disposition in the M+ LTR for the non-limiting events topic is applicable to PBAPS. Specifically, the licensee addressed the following non-limiting anticipated operational occurrences (AOOs) as follows:

- Slow Recirculation Increase: [[ ]]
- Fast Recirculation Increase: [[ ]]
- Generator Load Rejection: [[ ]]

- Main Steam Isolation Valve (MSIV) Closure, All Valves: [[

  ]]

- MSIV Closure, One Valve: [[

  ]]

- Turbine Trip, Bypass Failure, with Scram on High Flux (Failure of Direct Scram): [[

  ]]

- Loss of Feedwater Flow: [[

  ]]

The NRC staff concludes that the generic M+ LTR dispositions are applicable to PBAPS
[[
]]

SAR Section 9.2.1.3 - Main Steam Line Break Accident (Outside Containment)

As discussed in SAR Section 9.2.1.3, the licensee confirmed that the generic disposition in the M+ LTR for the main steam line break (MSLB) accident outside containment topic is applicable to PBAPS. Specifically, the source terms for the MSLB accident are dependent on the relative amount of water and steam released. Under MELLLA+ operating conditions, there will be an increase in steam and a decrease in water. This will result in lower releases such that the current analysis is bounding.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the current analysis for the MSLB accident is bounding.

SAR Section 9.2.1:4 - Loss-of-Coolant Accident (Inside Containment)

As discussed in SAR Section 9.2.1.4, the licensee confirmed that the generic disposition in the M+ LTR for the loss-of-coolant accident (LOCA) inside containment topic is applicable to PBAPS. Specifically, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the LOCA analysis is unaffected by operation in the MELLLA+ operating domain.

SAR Section 9.2.1.7 - Fuel Handling Accident

As discussed in SAR Section 9.2.1.7, the licensee confirmed that the generic disposition in the M+ LTR for the fuel-handling accident (FHA) topic is applicable to PBAPS. Specifically, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the FHA under MELLLA+ operating condition is bounded by the FHA analysis for current plant operating conditions.

SAR Section 9.3.2 - Station Blackout

As discussed in SAR Section 9.3.2, the licensee confirmed that the generic disposition in the M+ LTR for the station blackout (SBO) topic is applicable to PBAPS. Specifically, implementing MELLLA+ does not change the reactor power level, decay heat, or reactor operating pressure. In addition, there are no significant changes in the MS flow rate. Therefore, [[

]]

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

SAR Section 9.3.3 – Anticipated Transients without Scram with Core Instability

As shown in the table in SAR Section 9.3, the anticipated transients without scram (ATWS) with core instability (ATWSI) topic was generically dispositioned in the M+ LTR. However, plant-specific evaluation was performed for PBAPS. As such, the ATWSI topic is discussed in Section 3.3.9 of this SE.

3.2.10 SAR Section 10.0 - Other Evaluations

The following is a brief summary of the licensee's generic MELLLA+ dispositions for PBAPS for the topics in Section 10.0 of the SAR.

SAR Section 10.1.1 - High Energy Line Break - Steam Lines

As discussed in SAR Section 10.1.1, the licensee confirmed that the generic disposition in the M+ LTR for the high energy line break (HELB) steam lines topic is applicable to PBAPS. Specifically, a review of the heat balances produced for PBAPS MELLLA+ operation confirms there is no effect on the steam pressure or enthalpy at the postulated break locations (e.g., MS, HPCI, and RCIC).

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

SAR Section 10.1.2 - High Energy Line Break - Balance-of-Plant Liquid Lines

As discussed in SAR 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 PBAPS. Specifically, a review of the heat balances produced for PBAPS MELLLA+ operation confirmed there is no effect on the liquid line conditions at the postulated FW, RWCU, and RHR break locations. In addition, the

mass and energy release for operation in the MELLLA+ domain is bounded by the MELLLA domain analyzed for EPU, including final FW temperature reduction (FFWTR).

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

SAR Section 10.1.3 - High Energy Line Break - Other Liquid Lines

As discussed in SAR 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 PBAPS. Specifically, a review of the PBAPS design basis confirms that there are no additional high energy lines beyond those covered by SAR Section 10.1.2.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because there are no other high energy lines that require evaluation.

SAR Section 10.2.1 - Moderate Energy Line Break - Flooding

As discussed in SAR Section 10.2.1, the licensee confirmed that the generic disposition in the M+ LTR for the moderate energy line break (MELB) flooding topic is applicable to PBAPS. Specifically, a review of the PBAPS auxiliary flow rates and system inventories shows the MELLLA+ operating domain expansion does not affect the flow rates of moderate energy piping systems. In addition, for PBAPS, no operational modes evaluated for MELB are affected by the MELLLA+ operating domain expansion. [[

]]

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

SAR Section 10.2.2 - Moderate Energy Line Break - Environmental Qualification

As discussed in SAR Section 10.2.2, the licensee confirmed that the generic disposition in the M+ LTR for the MELB environmental qualification (EQ) topic is applicable to PBAPS. Specifically, a review of the PBAPS auxiliary flow rates and system inventories shows that the MELLLA+ operating domain expansion does not affect the flow rates of moderate energy piping systems. In addition, for PBAPS, no operational modes evaluated for MELB are affected by the MELLLA+ operating domain expansion. [[

]]

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

]]

SAR Section 10.3.1 - Electrical Equipment Environmental Qualification

As discussed in SAR Section 10.3.1, the licensee confirmed that the generic disposition in the M+ LTR for the electrical equipment environmental qualification (EQ) topic is applicable to

PBAPS. Specifically, for PBAPS under MELLLA+ operating conditions, there is no change in reactor power, radiation levels, decay heat, reactor operating pressure, MS flow rate, or FW flow rate. In addition, [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the EQ of electrical equipment [[ ]]

SAR Section 10.3.2 - Mechanical Equipment with Non-Metallic Components Environmental Qualification

As discussed in SAR Section 10.3.2, the licensee confirmed that the generic disposition in the M+ LTR for the mechanical equipment with non-metallic components EQ topic is applicable to PBAPS. Specifically, implementing MELLLA+ does not change the normal process temperatures or radiation levels in any of the plant areas where safety-related equipment is located. [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because the EQ of mechanical equipment with non-metallic components [[

]]

SAR Section 10.3.3 - Mechanical Component Design Qualification

As discussed in SAR Section 10.3.3, the licensee confirmed that the generic disposition in the M+ LTR for the mechanical component design qualification topic is applicable to PBAPS. Specifically, implementation of MELLLA+ does not change normal process temperatures, pressures, and flow rates. In addition, there is no change in radiation levels in any of the plant areas where safety-related equipment is located. [[

]]

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because mechanical component design qualification [[

]]

SAR Section 10.7.2 - Flow-Accelerated Corrosion

As discussed in SAR Section 10.7.2, the licensee confirmed that the generic disposition in the M+ LTR for the flow-accelerated corrosion (FAC) topic is applicable to PBAPS. Specifically, for PBAPS, 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 SE Section 3.3.3 (under SAR Section 3.3.3), the MCO values under MELLLA+ conditions are bounded by the pre-MELLLA+ conditions.

For PBAPS, 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" (Reference 32). The requirements of GL 89-08 are implemented at PBAPS by utilization of the Electric Power Research Institute (EPRI) generic program, CHECWORKS™. PBAPS-specific parameters are entered into this program to develop requirements for monitoring and maintenance of specific system components. No changes are required to the PBAPS-specific parameters that are entered into the CHECWORKS™ program as a result of MELLLA+ operation. In addition, the Maintenance Rule provides oversight for other mechanical and electrical equipment important to safety, to monitor performance and guard against age-related degradation.

The NRC staff concludes that the generic M+ LTR disposition is applicable to PBAPS because FAC, under MELLLA+ operating conditions, is bounded by current plant operation.

SAR Section 10.8 - NRC and Industry Communications

As discussed in SAR Section 10.8, the licensee confirmed that the generic disposition in the M+ LTR for the NRC and industry communications topic is applicable to PBAPS. 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 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 inherently include any effect as a result of NRC and industry communications, it is not necessary to review prior communications as part of the MELLLA+ review.

The NRC staff concludes that the generic M+ LTR disposition is applicable since the generic disposition indicates that no additional information needs to be provided for a MELLLA+ application in this area.

3.2.11 Generic MELLLA+ Dispositions Conclusion

The NRC staff concludes that for the generic dispositions discussed in SE Sections 3.2.2 through 3.2.10, operation in the MELLLA+ operating domain is acceptable, since the dispositions were consistent with the criteria discussed in SE Section 3.2.1. Specifically, these topics were dispositioned (in general) by either (1) providing or referencing a bounding analysis for the limiting conditions; (2) demonstrating that there is negligible effect due to MELLLA+ operation; (3) identifying the portions of the plant that are unaffected by the MELLLA+ power/flow map operating domain expansion; or (4) demonstrating that the sensitivity to MELLLA+ is small enough that the required plant-specific reload process is sufficient and appropriate for establishing the MELLLA+ licensing basis.

3.3 Plant-Specific MELLLA+ Dispositions

3.3.1 Introduction

As discussed in SE Section 3.1 above, the SAR provides a systematic disposition of the M+ LTR topics applied to PBAPS, including performance of plant-specific assessments and

confirmation of the applicability of those topics that were generically dispositioned in the M+ LTR.

The NRC staff evaluation of the licensee's MELLLA+ plant-specific dispositions for PBAPS is provided in SE Sections 3.3.2 through 3.3.10. Section 3.3.11 of this SE provides the staff's conclusion of the licensee's plant-specific MELLLA+ dispositions.

3.3.2 SAR Section 2.0 - Reactor Core and Fuel Performance

Section 2.0 of the SAR, "Reactor Core and Fuel Performance," indicates that each of the topics covered in this section was generically dispositioned in the M+ LTR. As such, no plant-specific assessments were required by the licensee for PBAPS. The NRC staff evaluation of the generic disposition of the reactor core and fuel performance topics is provided in Section 3.2.2 of this SE.

3.3.3 SAR Section 3.0 - Reactor Coolant and Connected Systems

The following provides the NRC staff's evaluation of the plant-specific assessments for the topics in Section 3.0 of the SAR.

SAR Section 3.1.2 - Overpressure Relief Capacity

The licensee's plant-specific assessment states that for PBAPS, the limiting overpressure event is the main steam isolation valve (MSIV) closure with scram on high flux (MSIVF). The peak reactor pressure vessel (RPV) bottom head pressure remains less than the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) allowable peak pressure of 1,375 psig (i.e., 110 percent of the reactor design pressure of 1,250 psig).

The licensee stated that the AOO, ASME overpressure, and ATWS response evaluations for MELLLA+ are performed using the same PBAPS safety relief valve (SRV) and spring safety valve (SSV) setpoint tolerance (i.e., 3 percent) used in the current PBAPS analysis. There are no changes to the PBAPS current licensing basis assumptions and code inputs used for the PBAPS ASME overpressure event.

The licensee stated that its analysis of the limiting overpressure event for PBAPS demonstrates that no change in overpressure relief capacity is required. In addition, the ATWS analysis concludes that no increase in the number of SRVs credited in the analysis is required to demonstrate acceptable results. Furthermore, no other changes in the pressure relief system or SRV and SSV setpoints are required for MELLLA+ operation. The ASME overpressure event will continue to be analyzed as part of each reload analysis and reported in the SRLR.

The NRC staff concludes that the licensee has adequately evaluated the proposed MELLLA+ operation with respect to overpressure relief capacity and has demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded.

SAR Section 3.2.1 - Fracture Toughness

The licensee's plant-specific assessment states that MELLLA+ operation results in slightly higher neutron flux in the upper part of the core due to decreased density. This results in [[ ]] in peak vessel and peak shroud flux. The licensee stated that the change in peak fluence is [[ ]]

The licensee stated that since there is negligible change to the peak fluence, there is also negligible change to the beltline adjusted reference temperature (ART). Therefore, the current pressure-temperature curves remain bounding for MELLLA+ operation. In addition, the upper shelf energy (USE) will maintain the margin requirements of Appendix G to 10 CFR Part 50.

Based on the above, the NRC staff concludes that the impacts of MELLLA+ operation with respect to fracture toughness are acceptable.

SAR Section 3.3.1.3 - Reactor Internal Pressure Differences (Acoustic and Flow-Induced Loads) for Faulted Conditions

The licensee's plant-specific assessment states that the loads in the RPV annulus on the jet pumps, core shroud, and core shroud support are not increased as a result of the MELLLA+ operating domain expansion. Specifically, [[ ]]

Based on the above, the NRC staff concludes that the impacts of MELLLA+ operation with respect to reactor internal pressure differences are acceptable.

SAR Section 3.3.2 - Reactor Internals Structural Evaluation

The licensee's plant-specific assessment states that [[ ]]

Based on the above, the NRC staff concludes that the impacts of MELLLA+ operation with respect to the reactor internals structural evaluation is acceptable.

SAR Section 3.3.3 - Steam Separator and Dryer Performance

The licensee provided its plant-specific assessment regarding steam separator and steam dryer performance in Section 3.3.3 of the SAR, as supplemented by its letters dated July 6, 2015 (Reference 5), and September 4, 2015 (Reference 6).

By Amendment Nos. 293 and 296 (Units 2 and 3, respectively) dated August 25, 2014 (ADAMS Accession No. ML14133A046), the NRC approved a 12.4 percent extended power uprate (EPU) that authorized an increase in the maximum thermal power level from 3514 MWt to the CLTP level of 3951 MWt. As part of the plant modifications for the EPU, the licensee replaced the steam dryer in each of the units. Section 2.2.6 of the NRC staff's SE for the EPU evaluated the structural integrity of the replacement steam dryers (RSDs) under EPU operating conditions. The following discussion provides the NRC staff evaluation regarding the structural integrity of the RSDs for EPU conditions combined with MELLLA+ conditions.

As discussed above in SE Section 3.2.8, the moisture content of the MS leaving the vessel may increase while operating near the minimum core flow in the MELLLA+ operating domain. The amount of liquid water carried in the steam is called moisture carry-over (MCO). As discussed in the licensee's letter dated July 6, 2015, due to lower core flow conditions at MELLLA+ as compared to EPU: (1) steam quality entering the steam separator increases; (2) maximum MCO exiting the steam separator and entering the RSD decreases; and (3) MCO exiting the RSD decreases. Therefore, the licensee concluded that the EPU MCO analysis is bounding for the combined EPU and MELLLA+ conditions for the RSD. The EPU MCO bounding analysis uses an MCO value of 0.3 weight percent. This value is conservative since the RSD is designed for an MCO value of 0.1 weight percent (at the exit of the RSD).

To address the impact on the RSD structural analysis for EPU conditions combined with MELLLA+ conditions, [[

]]

[[

]] These biases were added to the EPU RSD structural analysis results to support operation in the MELLLA+ operating domain for both Units 2 and 3.

The PBAPS, Units 2 and 3, RSD stress analyses performed for EPU operation included contributions from recirculation pump vane passing frequency effects that covered the expanded MELLLA+ domain of 83 percent to 110 percent of rated core flow.

The licensee determined that additional bias to account for the effect of expanded core flow domain associated with MELLLA+ conditions was necessary. This additional bias, which is applied to the upper and lower portions of the RSD analysis results at EPU, was determined based on the Unit 2 on-dryer strain gage measurements at different core flows. The NRC staff finds that this methodology to adjust RSD stresses at EPU conditions for PBAPS, Units 2 and 3, to capture the core flow effects in the MELLLA+ extended domain is reasonable. The deadweight, reactor internal pressure differences, seismic, and SRV loads remain the same or are bounded by those considered in the EPU evaluation. The licensee demonstrated that the high cycle fatigue stresses in steam dryer from flow-induced vibration, as well as for ASME Levels A, B, C, and D meet the respective allowable stresses. The licensee's analysis, as shown in its letter dated September 4, 2015, also demonstrated that the final minimum alternating stress ratio (MASR) values for Units 2 and 3 for combined EPU and MELLLA+ operation meet the applicable allowable limits (i.e., MASRs will be greater than 1.0).

The NRC staff concludes that the proposed license amendment to operate PBAPS, Units 2 and 3, at EPU conditions combined with MELLLA+ conditions for the RSDs is acceptable with respect to potential adverse flow effects for high-cycle fatigue, as well as to withstand the ASME normal, upset, emergency, and faulted load combinations. The NRC staff also concludes that

the RSDs will maintain their structural integrity for the combined EPU and MELLLA+ flow conditions.

3.3.4 SAR Section 4.0 - Engineered Safety Features

The following provides the NRC staff's evaluation of the plant-specific assessments for the topics in Section 4.0 of the SAR.

SAR Section 4.1.1 - Short-Term Pressure and Temperature Response

The M+ LTR states that operation in the MELLLA+ range may change the break energy for the design-basis accident (DBA) recirculation suction line break (RSLB). [[

]]

In its letter dated February 6, 2015 (Reference 3), the licensee describes that the limiting event for determination of peak drywell pressure (and also for evaluation of the containment LOCA hydrodynamic loads) is the double-ended guillotine break (DEGB) RSLB for the MELLLA+ analysis. It is noted that a change in the break subcooling associated with MELLLA+ operation can potentially affect the critical liquid break flow rate that controls the DEGB RSLB drywell pressure and temperature response during the period when peak drywell pressures occur. The RSLB is limiting relative to the main steam line break (MSLB) for Mark I plants. There are no equipment out-of-service (EOOS) options associated with the MELLLA+ RSLB LOCA containment analysis. Analyses of the short-term containment response to this event were performed at two statepoints in the MELLLA+ operating domain, including the 102.0 percent power, 83.0 percent core flow condition and the 80.8 percent power, 55 percent core flow condition that correspond to Points J and K in Figure 1-1 of the SAR, respectively. The results determined that Point J (102 percent power, 83.0 percent core flow) is the more limiting point of the MELLLA+ domain.

Table 4-1 of the SAR shows the comparison of the peak containment pressure for the limiting Point J to the limiting cases for CLTP for the design and bounding cases. The difference between the design and bounding cases is the initial containment conditions as shown in the last column of Table 4-1. Table 4-1 shows that the peak RSLB pressures for the MELLLA+ operating domain are bounded by peak pressures obtained for the CLTP RSLB and are below the design limit of 56.0 psig.

Table 4-1 of the SAR shows that the calculated peak drywell pressure for the bounding case (48.7 psig) obtained for the CLTP RSLB (EPU conditions) is below the current primary containment leakage testing pressure (Pa) value of 49.1 psig stated in PBAPS TS 5.5.12. Therefore, the licensee is not required to change the Pa value in the TSs.

[[

]] The results obtained for the spectrum of steam break analyses that were performed for the EPU are also bounding for the

MELLLA+ domain as shown in Table 1-1 of Attachment 4 to the licensee's letter dated February 6, 2015.

These comparisons of MELLLA+ results to CLTP results demonstrate that the drywell pressure and temperature responses in the MELLLA+ operating domain are bounded by the CLTP results. Therefore, the NRC staff finds the licensee's plant-specific assessment is acceptable.

SAR Section 4.1.2 - Containment Dynamic Loads

The M+ LTR requires a plant-specific evaluation to determine the effect of MELLLA+ operating domain expansion on the LOCA containment dynamic loads. These loads include LOCA loads and subcompartment pressurization. Details for this evaluation are presented below.

SAR Section 4.1.2.1 - LOCA Loads

The LOCA dynamic loads include vent thrust, pool swell, condensation oscillation, and chugging loads. These loads have been defined generically for Mark I plants as part of the Mark I containment program and are described in detail in the Mark I Containment Load Definition Report (LDR) which was approved by the NRC in NUREG-0661 (Reference 33).

The licensee used the results from [[ ]] to evaluate the effect of the MELLLA+ operating domain expansion on LOCA containment dynamic loads. The key parameters for this evaluation are [[ ]] The licensee's evaluation showed that these loads in the current operating domain bound the same loads in the MELLLA+ domain.

The NRC staff finds the licensee's evaluation reasonable and acceptable because the current containment pressure and temperature responses remain bounding.

SAR Section 4.1.2.2 - Subcompartment Pressurization

An annular structure of reinforced concrete is located inside the drywell around the reactor pressure vessel (RPV) in order to provide thermal and radiation shielding and is called the sacrificial shield wall (SSW). The SSW is designed to withstand the differential pressure (DP) or pressure difference that would develop across the wall as a result of a high pressure pipe break within the annulus (i.e., between the RPV and the SSW).

The licensee stated that there is a potential for the liquid line breaks at MELLLA+ operation leading to increased break flow rates, and thus, requiring plant-specific subcompartment pressurization evaluations as performed for PBAPS. The plant-specific evaluation was performed by the licensee and presented in the SAR. The staff's evaluation is provided below.

Subcompartment Pressurization for SSW

To determine the maximum pressure difference on the SSW due to the limiting RSLB between the RPV and the SSW, the licensee treated the break flow as a subcooled liquid mass release. The results, including the effects of the CLTP and the limiting off-rated condition, along the MELLLA operating domain upper boundary (minimum recirculation pump speed (MPS) point with FFWTR), in addition to the MELLLA+ operating domain, indicate that the design limit of the

SSW pressure difference is not exceeded for the whole operating domain, including the MELLLA+ domain. The results of the analysis are shown in SAR Table 4-2.

The NRC staff finds the licensee's evaluation acceptable because the results show that the SSW pressure difference design limit is not exceeded under MELLLA+ conditions.

Subcompartment Pressurization for Shield Plugs

To determine the maximum pressure difference on the SSW shield plugs due to the FW line break pressurizing the shield annulus, the licensee considered [[ ]] In addition, the thrust multiplication factor for jet impingement on the shield plugs is applied. The results, including the effects of the CLTP and the limiting off-rated condition along the MELLLA operating domain upper boundary (MPS point with FFWTR), in addition to the MELLLA+ operating domain, indicate that the design limit of the SSW shield plug pressure difference is not exceeded for the entire operating domain as resulting from the limiting FW line break at CLTP and MELLLA+ conditions. The results of the analysis are shown in SAR Table 4-3.

The NRC staff finds the licensee's evaluation acceptable because the results show that the SSW shield plug pressure difference design limit is not exceeded under MELLLA+ conditions.

SAR Section 4.2.6.1 – ECCS NPSH During ATWS

Consistent with the M+ LTR Limitations and Conditions 12.23.9 and 12.23.10, the licensee performed a plant-specific evaluation of ECCS pump NPSH for ATWS. The plant-specific MELLLA+ ATWS analysis, in Section 9.3 of the SAR, shows an increase in the peak suppression pool (torus) temperature from the current licensing basis peak suppression pool temperature. For the MELLLA+ ATWS event, the only ECCS pumps operating from the suppression pool are the RHR pumps. The HPCI pumps supply makeup to the RPV with alignment to the condensate storage tank (CST), which is unchanged from the PBAPS current licensing basis for ATWS. The core spray (CS) pumps are not credited for the ATWS event, which is also consistent with the PBAPS current licensing basis. Therefore, only the RHR pump NPSH is evaluated by the licensee for the MELLLA+ ATWS event. The NPSH margin for the RHR pumps is evaluated for the limiting conditions following an ATWS. The limiting NPSH conditions depend on the pump flow rates, debris loading on the suction strainers (for debris generating events), pipe frictional losses, suppression pool level, and suppression pool temperature.

Consistent with the PBAPS current licensing basis, maximum torus pressure is assumed to be 14.638 psia and no containment accident pressure (CAP) is used for calculating net positive suction head available (NPSHA). MELLLA+ calculations for the RHR pump NPSH are consistent with RG 1.82 (Reference 19) for the DBA-LOCA and meet the requirements of NRC guidance in Enclosure 1 of SECY 11-0014 (ADAMS Accession No. ML102780586) with respect to CAP. The torus water level for the ATWS event NPSH analysis is adjusted for the drawdown level consistent with RG 1.82 requirements. For ATWS, torus water level will increase from the initial torus water level, due to the use of HPCI for RPV makeup with the HPCI inventory being supplied from the CST. Therefore, for the NPSH analysis of ATWS, inventory addition from the CST is credited in the torus water level calculation.

ATWS is a non-LOCA event. Consistent with the guidance and the current licensing basis for PBAPS, the licensee calculated the ECCS pump required NPSH (NPSHR) used in NPSH margin evaluations at MELLLA+ conditions. The NPSHR contains an uncertainty for the large break DBA-LOCA and small break LOCA, and none for other events. The NPSH margin was calculated assuming a system flow rate that meets or exceeds the RHR pump operational requirements for ATWS. Consideration of ECCS suction strainer debris loading within the NPSH evaluations at MELLLA+ conditions is consistent with the PBAPS current analysis of record for the large break (RSLB) DBA-LOCA event. For PBAPS MELLLA+ operation, the ATWS event includes RHR pump suction strainer debris loading in the NPSH evaluation, which is consistent with the current licensing basis. The limiting values of the maximum suppression pool temperature, available NPSH (NPSHA), NPSH margin, and operating time in the maximum erosion zone for the ATWS event were calculated. The NPSH analysis results for the ATWS event are conservative with respect to maximum suppression pool temperature. The pump flow rates used in the ECCS NPSH evaluation are conservatively higher than those used in the safety analysis that provides the suppression pool temperature response. The RHR pumps have been analyzed for plant-specific conditions and have sufficient NPSH margin to perform satisfactorily during an ATWS initiated under MELLLA+ conditions. This plant-specific analysis is consistent with M+ LTR Limitation and Condition 12.17 concerning evaluation of the safety system performance during the long-term cooling phase of an ATWS in terms of NPSHA. Therefore, PBAPS meets all M+ LTR dispositions for the ECCS NPSH.

The NRC staff finds the licensee's evaluation acceptable because the required plant-specific evaluation has been conservatively performed and the results demonstrate that the ECCS pumps have sufficient margin during an ATWS under MELLLA+ conditions.

SAR Section 4.3.2 - Large Break Peak Clad Temperature

The large break LOCA evaluation, discussed in SAR Sections 4.3.1 and 4.3.2, concludes that MELLLA+ operation primarily affects the first increase in peak cladding temperature (PCT); therefore, the limiting single failure is not affected by MELLLA+ operation and remains the battery single failure.

[[

]] Mid-peak power shapes are the most limiting but produce acceptable values.

The NRC staff finds the licensee's evaluation acceptable because the PCT under MELLLA+ operating conditions remains below the regulatory limit of 2200 °F.

SAR Section 4.3.3 - Small Break Peak Clad Temperature

As discussed in SAR Sections 4.3.1 and 4.3.3, the licensee performed calculations for a spectrum of break sizes for the small break LOCA at MELLLA+ conditions. [[

]] The Cycle 21 SRLR (Reference 2) concludes that the maximum Appendix K PCT is 1910 °F at EPU power and MELLLA+ core flow, while Appendix K PCT is 1905 °F at EPU power and rated core flow.

The NRC staff finds the licensee's evaluation acceptable because the PCT under MELLLA+ operating conditions remains below the regulatory limit of 2200 °F.

3.3.5 SAR Section 5.0 - Instrumentation and Control

The following provides the NRC staff's evaluation of the plant-specific assessments for the topics in Section 5.0 of the SAR.

SAR Section 5.3.1 - APRM Flow-Biased Simulated Thermal Power Scram

As discussed in SAR Section 5.3.1, the analytical limit (AL) for the APRM flow-biased simulated thermal power scram function is established to [[

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The NRC staff concludes that the licensee's plant-specific evaluation is consistent with the methods described for this topic in the M+ LTR, and therefore, is acceptable.

The NRC staff's evaluation of the TS changes associated with the APRM flow-biased simulated thermal power scram is discussed in SE Section 4.2.

3.3.6 SAR Section 6.0 - Electrical Power and Auxiliary Systems

The following provides the NRC staff's evaluation of the plant-specific assessments for the topics in Section 6.0 of the SAR.

SAR Section 6.3.4.1 - New and Spent Fuel Storage Criticality Review

As discussed in Appendix C to the NRC staff SE for the M+ LTR, the plant-specific application should include confirmation or discussion on how the spent fuel criticality requirement can be met for bundles that operated at MELLLA+ conditions.

As discussed in SAR Section 6.3.4.1, the licensee stated that the spent fuel storage criticality analyses include conservative assumptions relative to enrichment, exposure, and void history. The licensee further stated that its analyses show margin to fuel storage criticality safety limits and ensure that fuel storage racks will maintain sub-critical conditions in the SFP. In addition, the CLTP SFP criticality analysis with GNF2 fuel remains applicable for operation in the MELLLA+ region.

Based on the above, the NRC staff concludes that the licensee's plant-specific evaluation is acceptable.

SAR Section 6.5.2 - SLC System Hardware

As discussed in SAR Section 6.5.2, the PBAPS reactor operating pressure is unchanged by the MELLLA+ operating domain expansion. In addition, there are no changes to the PBAPS SRV or SSV setpoints as a result of MELLLA+ operation. As such, the NRC staff concludes that the

SLC system will remain capable of performing its shutdown function under MELLLA+ conditions.

SAR Section 6.5.3 - SLC System ATWS Requirements

As discussed in the M+ LTR, the SLC system is typically designed for injection at a maximum reactor pressure equal to the upper analytical setpoint for the lowest group of SRVs operating in the relief mode. [[

]]

As discussed in SAR Section 6.5.3, the PBAPS plant-specific analysis shows that the maximum reactor lower plenum pressure, following the most limiting ATWS event, reaches 1,191 psig during the time the SLC system is analyzed to be in operation. There is no increase in the limiting ATWS results for CLTP within the MELLLA+ domain. The pressure margin for the pump discharge relief valves is 184 psig under MELLLA+ conditions. As such, the SLC system relief valves will remain closed during SLC system injection.

As also discussed in SAR Section 6.5.3, for the PBAPS loss of offsite power (LOOP) ATWS event, there is no difference in SRV/SSV operating characteristics for the bounding ATWS event in terms of peak reactor pressure. The minimum reactor pressure, just prior to when SLC initiates, remains low enough to ensure SLC system relief valve closure prior to the analyzed SLC initiation time in the event of an early initiation of SLC during the initial ATWS transient pressure response.

Based on the above, the NRC staff concludes that the licensee's plant-specific evaluation has adequately addressed the issues in the M+ LTR, and therefore, is acceptable.

3.3.7 SAR Section 7.0 - Power Conversion Systems

Section 7.0 of the SAR, "Power Conversion Systems," indicates that each of the topics covered in this section was generically dispositioned in the M+ LTR. As such, no plant-specific assessments were required for PBAPS. The NRC staff evaluation of the generic disposition of the power conversion systems topics is provided in Section 3.2.7 of this SE.

3.3.8 SAR Section 8.0 - Radwaste Systems and Radiation Sources

The following provides the NRC staff's evaluation of the plant-specific assessments for the topics in Section 8.0 of the SAR.

SAR Section 8.4.2 - Fission and Activated Corrosion Products

The MELLLA+ operating domain expansion does not involve a change in the current licensed maximum reactor thermal power nor the maximum rated reactor steam flow. During power operation, the radiation sources in the core are directly related to the fission rate. These sources include radiation from the fission process, accumulated fission products, and neutron activation of reactor components. Since the fission rate in the core is directly related to the

power output, there is no impact on these radiation sources from operating in the MELLLA+ operating domain.

The moisture content of the main steam (MS) leaving the vessel may increase while operating near the minimum core flow (CF) in the MELLLA+ operating domain. The amount of liquid water carried in the steam is called moisture carry-over (MCO) and can have a negative effect on turbine performance. Higher MCO will also result in an increase in the soluble and non-soluble radioactive species in the reactor water being transported from the reactor vessel to the turbine and secondary side of the plant. However, as discussed in SE Section 3.3.3 (under SAR Section 3.3.3), the MCO values under MELLLA+ conditions are bounded by the pre-MELLLA+ conditions. Therefore, the resulting MCO and radiation levels will remain within those currently allowed.

Based on the above, the NRC staff concludes that the fission and activated corrosion products under MELLLA+ operating conditions will be bounded by current plant operating conditions.

SAR Section 8.5.1 - Normal Operational Radiation Levels

The small increase in MCO from periodically operating at or near the MELLLA+ minimum CF rate may increase the deposition of non-volatile fission products, actinides and corrosion, and wear products from the reactor coolant onto the wetted surfaces of the turbine, condensate, and feed systems. Although the MCO values under MELLLA+ conditions are bounded by the pre-MELLLA+ conditions, the corresponding increase in dose rates associated with these deposited materials may be an additional source of occupational exposure during the repair and maintenance of these systems. However, the current as low as reasonably achievable (ALARA) program practices at PBAPS (e.g., work planning, source term minimization, etc.), coupled with existing radiation exposure procedural controls, will be able to compensate for any small increases in dose rates associated with MELLLA+ operations. Therefore, the NRC staff concludes that increased radiation sources resulting from this proposed MELLLA+ operating domain expansion, as discussed above, will not adversely impact the licensee's ability to maintain normal operational radiation doses within the applicable limits in 10 CFR Part 20 and ALARA.

SAR Section 8.5.2 - Post-Shutdown Radiation Levels

As discussed in SAR Section 8.5.2, the shutdown radiation levels are dominated by the accumulated contamination of some fission and activated corrosion products. These radionuclide concentrations in the reactor coolant do not vary significantly unless the MCO from the vessel increases. As discussed above, there may be a small increase in MCO from periodically operating at or near the MELLLA+ minimum CF rate. However, the MCO values under MELLLA+ conditions are bounded by the pre-MELLLA+ conditions. In addition, the current ALARA program practices at PBAPS (e.g., work planning, source term minimization, etc.), coupled with existing radiation exposure procedural controls, will be able to compensate for any small increases in dose rates associated with MELLLA+ operations. Therefore, the NRC staff concludes that increased radiation sources resulting from this proposed MELLLA+ operating domain expansion, as discussed above, will not adversely impact the licensee's ability to maintain post-shutdown radiation doses within the applicable limits in 10 CFR Part 20 and ALARA.

SAR Section 8.5.3 - Post-Accident Radiation Levels

Post-accident radiation levels depend primarily upon the core inventory of fission products and TS levels of radionuclides in the coolant. The post-accident source term is similarly dependent on the maximum licensed power. Since there is no change to the maximum licensed power, operation in the MELLLA+ domain has no impact on the in-plant radiological hazards during an accident or on the licensee's assessment of vital area access per NUREG-0737, Item II.B.2 (Reference 18).

3.3.9 SAR Section 9.0 - Reactor Safety Performance Evaluations

The following provides the NRC staff's evaluation of the plant-specific assessments for the topics in Section 9.0 of the SAR.

SAR Section 9.1 - Anticipated Operational Occurrences

The licensee has performed a plant-specific evaluation of anticipated operational occurrence (AOO) events as discussed in SAR Sections 9.1.1 and 9.1.2. For PBAPS, the following AOOs were analyzed [[ ]]:

- Generator Load Rejection Without Bypass (LRNBP)
- Turbine Trip Without Bypass (TTNBP)
- FW Controller Failure (Maximum Demand) (FWCF)
- Inadvertent HPCI Start With Level 8 Trip (HPCIL8)
- Loss of FW Heater (LFWH)
- Rod Withdrawal Error (RWE)

These AOOs were evaluated at the current licensed thermal power (CLTP), which is equivalent to 120 percent of the original licensed thermal power (OLTP), and at two flows; the increased core flow (ICF) limit of 110 percent and the MELLLA+ reduced core flow limit of 83 percent. A summary of the analysis results is presented in Table 3.3.9-1 below.

The limiting events are LRNBP and TTNBP, which result in a relative change in critical power ratio (ΔCPR/ICPR) of 0.19. The limiting condition occurs at the low flow condition, which results in an increase of +0.03 in the uncorrected ΔCPR/ICPR for MELLLA+ operation.

Table 3.3.9-1 - Comparison of AOO Analyses Results at 83% and 110% Core Flow

Event	Parameter	Units	Results at 83%	Results at 110%
LRNBP	P (MWt)/CF (% rated)		3514/110	3951/83
	Peak Neutron Flux	% Initial	305	289
	Peak Heat Flux	% Initial	107	108
	Peak Vessel Pressure	psia	1260	1278
	GNF2 ΔCPR/ICPR	N/A	**0.16**	**0.19**
TTNBP	P (MWt)/CF (% rated)		3514/110	3951/83
	Peak Neutron Flux	% Initial	280	271
	Peak Heat Flux	% Initial	106	107
	Peak Vessel Pressure	psia	1259	1277
	GNF2 ΔCPR/ICPR	N/A	**0.16**	**0.19**
FWCF	P (MWt)/CF (% rated)		3514/110	3951/83
	Peak Neutron Flux	% Initial	194	190
	Peak Heat Flux	% Initial	110	109
	Peak Vessel Pressure	psia	1226	1245
	GNF2 ΔCPR/ICPR	N/A	**0.14**	**0.17**
HPCIL8	P (MWt)/CF (% rated)		3514/110	3951/83
	Peak Neutron Flux	% Initial	182	187
	Peak Heat Flux	% Initial	113	114
	Peak Vessel Pressure	psia	1224	1242
	GNF2 ΔCPR/ICPR	N/A	**0.14**	**0.18**
LFWH	P (MWt)/CF (% rated)		3951/99	3951/83
	GNF2 ΔCPR	N/A	**0.12**	**0.12**
RWE	P (MWt)/CF (% rated)		3951/100	3951/83
	GNF2 ΔCPR	N/A	**0.27**	**0. 27**

The operating limits to critical power ratio and linear heat generation rate (LHGR) are adjusted upwards when operating at off-nominal conditions by power-dependent and flow-dependent factors. The licensee has calculated the slow recirculation flow increase under MELLLA+ conditions to evaluate the flow-dependent limits for a representative MELLLA+ equilibrium core. The results of these analyses are discussed in Section 9.1.2 of the SAR. The NRC staff concludes that the results indicate that the existing PBAPS limits are adequate for MELLLA+ operation.

GNF2 Performance at Low Flows

The PBAPS SAR calculations are based on a full equilibrium core of GNF2 fuel. Even though GNF2 [[

]] On a typical two recirculation pump trip (2RPT), the CPR increases when the flow is reduced. Later in the transient, CPR is degraded if oscillations are established. The CPR increase due to the initial flow reduction tends to dominate the final results on the analysis. A similar effect can be observed on the AOO analyses for MELLLA+ operation. The low flow conditions (80 percent flow) tend to be limiting in these analyses, while the opposite is more common with fuels other than GNF2.

GEH simplified stability (GS-3) is a newly approved stability long-term solution methodology. The GS-3 methodology is intended to replace the Option I-D, II and III setpoint methodology (based on the old delta over initial MCPR versus oscillation magnitude) with [[ ]] During an NRC staff audit for the GS-3 methodology, the staff reviewed a number of calculations and observed that the [[ ]] For all the cases the staff reviewed, a similar trend was observed. Figure 3.3.9-1 below shows a calculation for the [[ ]]

Therefore, the NRC staff concludes that the MELLLA+ results documented in Table 3.3.9-1 are consistent with the expected [[ ]].

Figure 3.3.9-1 – [[ ]]

[[ ]]

Figure 3.3.9-2 – [[

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SAR Section 9.2.1 - Design-Basis Events

SAR Section 9.2.1 discusses the radiological consequences of design-basis events (DBAs). The radiological consequences of DBAs are evaluated to determine off-site doses, as well as control room operator doses. The only DBA with a plant-specific disposition applicable to PBAPS is a control rod drop accident (CRDA).

Core inventory source terms and the TS reactor coolant system source terms do not change because of MELLLA+ operation. There is no change in PBAPS licensed core power, decay heat, pressure, or steam flow as a result of the MELLLA+ operating range expansion.

As discussed in SAR Section 9.2.1.1, for PBAPS, two postulated CRDA events govern the analysis of radiological consequences. For Event 1, the release path is via the mechanical vacuum pump at low power operation. For Event 2, the release path is via the condenser and the steam jet air ejectors at normal power operation. The licensee indicated that 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. Because PBAPS may operate with portions of the off gas 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 alternate source term (AST) CRDA source term, and the peaking factor remains bounding. Since there are no changes to removal, transport, or dose conversion assumptions for this event, the PBAPS CRDA evaluation for the MELLLA+ operating domain is bounded by the analysis for the current licensed operating domain.

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

Since radiological consequences of a CRDA under MELLLA+ operating conditions is bounded by the analysis for current plant operation, the NRC staff concludes that the CRDA radiological consequences will remain below the design criteria specified in 10 CFR 50.67, "Accident source term," and the accident-specific design criteria outlined in RG 1.183.

SAR Section 9.3 - Special Events

Section 9.3 of the SAR discusses three special events: station blackout (SBO), anticipated transients without scram (ATWS), and ATWS with core instability (ATWSI). SBO was dispositioned generically and is discussed in SE Section 3.2.9. ATWS and ATWSI were dispositioned on a plant-specific basis and are discussed below.

SAR Sections 9.3.1 and 9.3.3 - ATWS and ATWSI

ATWS is defined as an AOO followed by the failure of the reactor protection system specified in draft GDC 14 and 15. The regulation at 10 CFR 50.62 requires, in part, that:

- Each BWR have an alternate rod insertion (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device.

- Each BWR have a SLC system with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gpm of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 (B-10) isotope abundance into a 251-inch inside diameter reactor vessel.

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

The NRC staff's review was conducted to ensure that (1) the above requirements are met; (2) sufficient margin is available in the setpoint for the SLC system pump discharge relief valve such that SLC operability is not affected by the proposed MELLLA+ operating domain expansion; and (3) operator actions specified in the plant's emergency operating procedures (EOPs) are consistent with the generic emergency procedure guidelines (EPGs) and severe accident guidelines (SAGs) insofar as they apply to the plant design. In addition, the NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME Service Level C limit of 1,500 psig; (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 °F; (3) the peak suppression pool temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design pressure. The NRC staff also evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the NRC staff reviewed the limiting event determination, the sequence of events,

the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses.

ATWS

The licensee analyzed the various ATWS events and concluded that the ATWS logic and setpoints remain unchanged for the proposed MELLLA+ operating domain expansion.
[[

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[[

]]

Two analysis methods are used: (1) the licensing basis methodology, which uses the ODYN computer code as the licensing basis for the PBAPS ATWS long-term analysis, and (2) a best estimate methodology, which uses the TRACG04 computer code with input data from the TGBLA06/PANAC11 computer codes as the licensing basis for the TRACG ATWS overpressure analysis. [[

]]

In accordance with M+ LTR Limitation and Condition 12.23.4, for PBAPS, [[

]]

For the licensing basis calculation [[

]]

With those assumptions, the peak vessel pressure is calculated [[
]] which is well below the 1,500 psig ASME Service Level C limit. [[
]] The calculations also show that MELLLA+ operation has [[ ]] on PCT and cladding oxidation because
[[ ]]

The [[ ]] calculation indicates that [[ ]] the suppression pool temperature would reach a temperature of [[ ]] which is below the pool design limit of 180 °F, which is below the NPSH limit for ECCS pumps and below the HCTL limit. The peak containment pressure is [[ ]] which is below the design limit of 56 psig.

PBAPS had increased the B-10 enrichment from 61.92 atom percent to 92 atom percent as part of the changes made due to the EPU. With the current PBAPS 92 percent B-10 enrichment (which is required by TS SR 3.1.7.10), the final suppression pool temperature is [[

]] Thus, the NRC staff concludes that the SLC system B-10 enrichment limits the integrated heat load to containment to a value comparable to the original design without MELLLA+ enhancements. In addition, since the conservative ODYN calculation shows that the HCTL is not reached, the best estimate ATWS calculation is not required for PBAPS.

Section 9.3.1 of the SAR presents the results of the ODYN ATWS analyses. For all cases analyzed, the following ATWS acceptance criteria are satisfied:

- Maintain reactor vessel integrity (i.e., the peak vessel bottom pressure is less than the ASME Service Level C limit of 1,500 psig).
- Maintain containment integrity (i.e., the peak suppression pool temperature is less than the design limit and the peak containment pressure is less than the containment design pressure).
- Maintain a coolable core geometry.

<u>ATWSI</u>

In addition, the licensee evaluated core instability during ATWS events, and the results are documented in Section 9.3.3 and in Table 9-5 of the SAR. The results of the ATWSI analysis show that the mitigation actions in the PBAPS EOPs (flow runback to uncover the spargers) are effective in the MELLLA+ operating domain. The TRACG04 calculations indicate that all applicable fuel limits are satisfied during these relatively small oscillations. The highest PCT during the most limiting ATWSI event was calculated to be [[ ]] which is significantly lower than the 10 CFR 50.46 limit of 2200 °F.

As part of the response to SRXB-RAI-12 in Reference 4, the licensee submitted a comparison of calculated PCT versus Tmin (minimum temperature for stable film boiling) during an ATWSI calculation. Middle of cycle (MOC) conditions are presented with a turbine trip with bypass (TTWBP) transient and failure to scram. As seen in the response to SRXB-RAI-12 (and discussed in Appendix A to this SE), the TTWBP cases, using the Modified Shumway Tmin correlation, [[ ]]

Based on information from independent test data relevant to ATWSI calculations, the NRC staff needed to learn more about the impact of modeling assumptions and correlations on ATWSI calculations. Specifically, based on this experimental data, the staff believes that there is uncertainty on the appropriateness of using the Modified Shumway correlation for Tmin during thermal-hydraulic instabilities, and this may yield non-conservative results. For this reason, the staff requested in SRXB-RAI-18 that sensitivity studies be performed using the Homogenous Nucleation Temperature (THN), which the staff believes to be conservative. The licensee provided the requested sensitivity calculations in a letter dated October 1, 2015 (Reference 27). Based on the NRC staff's evaluation of the sensitivities provided in the licensee's response to SRXB-RAI-18, the staff concludes that for the worst case sensitivities using THN and more

nominal assumptions, the ATWS acceptance criteria of maintaining a coolable core geometry are satisfied.

Based on the above, the NRC staff concludes that the PBAPS ATWS mitigation features [[ ]] are adequate to mitigate the ATWSI oscillations. The calculations indicate that ATWS acceptance criteria (i.e., maintain reactor vessel integrity, maintain containment integrity, and maintain coolable core geometry) are satisfied, even in the presence of unstable power oscillations.

<u>ATWS and ATWSI Conclusions</u>

The NRC staff has reviewed the information submitted by the licensee related to ATWS and ATWSI and concludes that the licensee has adequately accounted for the effects of the proposed MELLLA+ operating domain expansion on ATWS. Therefore, the NRC staff finds the proposed MELLLA+ operating domain expansion acceptable with respect to ATWS.

3.3.10 SAR Section 10.0 - Other Evaluations

The following provides the NRC staff's evaluation of the plant-specific assessments for the topics in Section 3.0 of the SAR.

SAR Section 10.4 - Testing

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

Section 10.4 of the SAR provides a brief description of plant-specific testing for implementation of the PBAPS MELLLA+ operating domain expansion. The testing proposed for PBAPS covers the full scope of testing described in the M+ LTR. 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 MCO magnitude and trend.

- 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 evaluate core thermal power, fuel thermal margin, and core flow performance to evaluate results against projected values and operational limits.

- A pressure regulator test will confirm that the pressure control system 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 setpoint step 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 confirm that plant operation is consistent with the analysis supporting the proposed MELLLA+ operating domain expansion.

SAR Section 10.5 - Individual Plant-Examination

As discussed in SAR Section 10.5, in accordance with the M+ LTR Limitation and Condition 12.21, a plant-specific probabilistic risk assessment (PRA) was performed by the licensee.

The NRC staff reviewed the PBAPS LAR and determined that it was not risk-informed but did provide risk insights related to the implementation of MELLLA+. Specifically, the licensee augmented the generic risk discussion contained in the M+ LTR with plant-specific information on initiating event frequencies, component reliability, operator response, success criteria, external events, shutdown risk, and PRA quality. The licensee reported an increase in core damage frequency (CDF) of 3.7×10^{-8} / year and an increase in large early release frequency (LERF) of 3.6×10^{-8} / year, primarily due to slight changes to human error probabilities associated with ATWS sequences.

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

SAR Section 10.6 - Operator Training and Human Factors

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

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 18 (Reference 23(q))

- NUREG-1764, "Guidance for the Review of Changes to Human Actions" (Reference 16)

- NUREG-0711, "Human Factors Engineering Program Review Model" (Reference 17)

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 performed 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 conflict with the licensee's assessment of risk importance or that of other portions of the NRC staff review. This assessment should not

be considered as an accurate assessment of risk when compared to other methods, especially those using plant-specific data and NRC-accepted methods of probabilistic risk analysis and human reliability analysis (PRA/HRA).

Description of Operator Actions Added, Changed, or Deleted

Section 2.4.4 of the SAR provides a basis for a new operator action to exit the MELLLA+ operating domain with a feedwater temperature reduction greater than 10 °F below the feedwater design temperature. There is no time limit associated with this in the MELLLA+ analysis; it is not a time critical action (TCA) in accordance with the PBAPS Operator Response Time Program.

Similarly, Section 1.2.4 of the SAR provides a basis for a new operator action to exit the MELLLA+ operating domain when operating in single loop operation. This is an immediate action and is not considered a TCA in accordance with the PBAPS Operator Response Time Program.

As discussed in the licensee's supplement dated February 6, 2015 (Reference 3), the following actions stated in Sections 9.3.1.1 and 9.3.3 of the SAR are current operator actions that are incorporated in procedures and training at PBAPS:

- Initiate SLC system boron injection
- Reduce reactor water level
- Initiate RHR suppression pool cooling

The above three operator actions are credited in the MELLLA+ licensing basis ATWS analysis and are identified as TCAs in accordance with the PBAPS Operator Response Time Program. The TCA associated with initiation of RHR suppression pool cooling at 660 seconds into an ATWS event was an existing TCA that was implemented as part of the PBAPS EPU. The new TCA response times assumed in the MELLLA+ analysis are as follows:

- SLC system pump initiation within 120 seconds
- Commence reactor water level reduction by reducing feedwater flow within 120 seconds

As discussed in the licensee's supplement dated October 1, 2015 (Reference 27), PBAPS completed training for the new TCAs supporting the MELLLA+ analysis for all five operating shift crews on August 14, 2015. All five crews were evaluated in order to assure that they can meet the new TCAs consistent with the assumptions in the MELLLA+ analysis. The average operator crew times were as follows:

- SLC system pump initiation: 73 seconds
- Commence reactor water level reduction: 84 seconds

The NRC staff has determined that the licensee has demonstrated reasonable assurance that the TCAs can be performed within the designated allowable time. Therefore, the staff has determined that this is acceptable for implementation of MELLLA+.

Operating Experience Review

Available operating experience related to MELLLA+ operation is limited because the first authorization to operate in the MELLLA+ domain was only recently issued (the Monticello Nuclear Generating Plant (MNGP) amendment dated March 28, 2014). Subsequent MELLLA+ amendments have been approved by the NRC for the Grand Gulf Nuclear Station, Unit 1 (GGNS), on August 31, 2015, and for the Nine Mile Point Nuclear Station, Unit 2 (NMP2), on September 2, 2015. As discussed in the supplement dated February 6, 2015, Exelon has been in contact with each of these plants to discuss implementation experience and various technical and operational questions that have arisen during development and review of the associated MELLLA+ amendments.

Exelon also performed searches for related MELLLA+ operating experience at the World Association of Nuclear Operators website and at the Institute of Nuclear Power Operations website. No specific information related to MELLLA+ operation was found on either website. The licensee performed an operating experience review with the only currently available information from other licensed facilities as noted above. Based on the lack of readily available information, the licensee is seeking information from its counterparts; therefore, the staff finds this acceptable.

Functional Requirements Analysis and Function Allocation

As discussed in the licensee's supplement dated February 6, 2015, PBAPS does not use the functional requirements analysis or function allocation to define design-basis operational requirements. The process governing changes and the addition of operator requirements is part of the Exelon configuration change control process at PBAPS. This process provides the necessary direction and guidance to evaluate configuration changes to the facility, including impact assessments that identify procedures and training material that require revisions for the planned configuration change.

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

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

Task Analysis

As discussed in the licensee's supplement dated February 6, 2015, based on the impact reviews conducted per Exelon's configuration change control process, new and changed tasks

will be identified and analyzed per the licensee's Nuclear Training Program and approved by the applicable line organizations. PBAPS employs a procedure-based task list for operations department duties. If new procedures are required, a new task will be established. Skills and knowledge required to perform new and changed tasks will be identified and form the basis for knowledge and performance objective development.

PBAPS will use an already existing configuration control process and Nuclear Training Program to identify the tasks that will be new or changed as a result of implementation of the MELLLA+ amendment. The NRC staff finds the licensee's treatment of this review element to be acceptable.

Staffing

As discussed in the licensee's supplement dated February 6, 2015, no new or additional staff is required, nor are there any new or additional qualifications required to perform the actions within the time constraints established. Operation in the MELLLA+ domain is not expected to increase operator workload. The NRC staff concludes that no additional staffing or qualifications or changes thereto are required; therefore, the NRC staff finds the licensee's treatment of this review element to be acceptable.

Probabilistic Risk and Human Reliability Analysis

As discussed above (SAR Section 10.5 - Individual Plant-Examination), the NRC staff reviewed the risk information provided by the licensee and concluded that the expected increase in risk associated with implementation of MELLLA+ at PBAPS would be well within the risk acceptance guidelines delineated by RG 1.174. Therefore, the NRC staff's review did not identify any "special circumstances" that would warrant an in-depth PRA review.

Human-System Interface Design

As discussed in the licensee's supplement dated February 6, 2015, in order to support the implementation of MELLLA+, an upgrade to the existing PRNM system at PBAPS will be made. This modification changes the human-system interfaces as follows:

- The main control room PRNM 2/4 logic module front panel has the additional confirmation density algorithm (CDA) trip indication and has updated trip nomenclature.
- The APRM interface is modified to provide controls for operators to enable ABSP.
- The operator display assembly is updated to include the CDA graph screens in the same manner as other PRNM graphs.
- There will be a new alarm for the CDA. It will use currently spare alarm points in the 2/4 logic module.
- The exit region alarm is incorporated as part of control room OPRM trip-enabled alarm.

- The existing OPRM pre-trip alarms for the growth rate based algorithm, and the amplitude based algorithm (ABA), are being eliminated. These pre-trip alarms are replaced by the CDA/period based detection algorithm pre-trip alarm.

Section 10.6 of the SAR describes that changes to automatic setpoints are implemented as design changes in accordance with the PBAPS configuration control procedures. The configuration control process includes a review by operations and training personnel. Training and implementation requirements are also identified and tracked, including effects on the simulator. The licensee uses training to ensure that operators are aware of changes to the control room and instrument/alarm setpoint changes.

Based on the above, the NRC staff finds the licensee's treatment of this review element to be acceptable.

Procedure Design

As discussed in the licensee's supplement dated February 6, 2015, changes to procedures will be developed in accordance with Exelon's configuration change control process, including impact reviews by operations and training personnel. Training and implementation requirements, including any effects on the simulator, are identified and tracked.

The NRC staff concludes that the normal licensee 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.

Training Program Design

As discussed in the licensee's supplement dated February 6, 2015, Exelon's "Systematic Approach to Training Process" encompasses training analysis, training material design and development, training implementation, and training effectiveness evaluation. This systematic training process is part of an overall set of integrated processes for the operation and support of Exelon's nuclear plants. The Exelon Nuclear Training Program provides specific direction and guidance on the performance of job and task analysis. Based on the impact reviews conducted as part of the configuration control change process, new and changed tasks will be identified and analyzed per the training program requirements and will be approved by the applicable line organizations. PBAPS employs a procedure-based task list for operations department duties. If new procedures are required, a new task will be established. Skills and knowledge required to perform new and changed tasks will be identified and will form the basis for knowledge and performance objective development. Required changes will be made consistent with the licensee's current training program requirements. These changes will be made consistent with similar changes made for other plant modifications and include any changes to the TS, EOPs, and plant systems.

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 this review element to be acceptable.

Human Factors Verification and Validation (V&V)

The PBAPS simulator has been updated to reflect the MELLLA+ analysis to support the implementation of the amendment. Additionally, procedure revisions will be completed as part of the implementation in accordance with the licensee's configuration change control process. As discussed above, operators have completed training associated with the TCAs supporting the MELLLA+ analysis. In addition, as discussed in the licensee's supplement dated February 6, 2015, the ATWS response procedures of other Exelon BWR plants were benchmarked. This benchmarking provided assurance that TCAs can be accomplished. The LAR stipulates that SLC actuation and feedwater flow reduction to mitigate ATWS can be completed in 120 seconds. During an audit by NRC staff at PBAPS in May 2015, the staff observed four ATWS scenarios for timing validation. Staffing levels were conservatively demonstrated at the minimum level of three operators. The demonstrations repeatedly showed the operators successfully initiating SLC at about 50 seconds and feedwater flow reduction at about 80 seconds into the event, leaving roughly a 70-second margin for SLC and 40-second margin for feedwater flow. This demonstration provides reasonable assurance that the actions are feasible within the time constraints.

The results of the MELLLA+ human factors review determined that changes to plant procedures will not alter the current mitigation strategies. Changes associated with setpoints will not introduce a level of complexity that would lead to misunderstanding the parameters.

Based on the impact reviews conducted per Exelon's configuration change control process, new and changed tasks will be identified and analyzed per the licensee's Nuclear Training Program and approved by the applicable line organizations. PBAPS employs a procedure-based task list for operations department duties. If new procedures are required, a new task will be established. Skills and knowledge required to perform new and changed tasks will be identified and will form the basis for knowledge and performance objective development.

As discussed above (in the section titled, "Description of Operator Actions Added, Changed or Deleted"), the licensee completed training of TCAs supporting the MELLLA+ analysis for all five operating shift crews on August 14, 2015. All five crews were evaluated in order to assure that they can meet the new TCAs consistent with the assumptions in the MELLLA+ analysis. Based on the results of this validation testing, the NRC staff concluded that the licensee has demonstrated reasonable assurance that the TCAs can be performed within the designated allowable time.

Based on the above, the NRC staff finds the licensee's treatment of this review element to be acceptable.

Human Performance Monitoring Strategy

As discussed in the licensee's supplement dated February 6, 2015, the operations director, or designee, is responsible for the operator response time program, including periodically validating that TCAs can be met. Operations training is responsible for developing and maintaining simulator scenario and walkthrough scenario validation materials and providing resources to support initial and periodic validations of TCAs. Operator response time program actions that have been selected for training are incorporated into the applicable initial and continuing training programs for equipment operators and licensed operators.

In addition, a senior management observation program provides guidance to senior line managers to conduct observations of control room crew performance. Observations are performed in the plant and in the simulator during training and/or evaluation scenarios.

The NRC staff finds the licensee's treatment of this review element to be acceptable because the existing program is adequate to monitor human performance.

Conclusion for SAR Section 10.6 - Operator Training and Human Factors

Based on the above, the NRC staff finds the proposed MELLLA+ amendment to be acceptable with respect to operator training and human factors.

SAR Section 10.9 - Emergency and Abnormal Operating Procedures

As discussed in SAR Section 10.9, EOPs and abnormal operating procedures (AOPs) can be affected by operating in the MELLLA+ domain. The EOPs include variables and limit curves, which define conditions where operator actions are indicated. The EOPs are symptom-based. AOPs include event-based operator actions.

The licensee stated that the EOPs and AOPs will be reviewed for any effect due to MELLLA+ operation and revised prior to MELLLA+ implementation. In addition, any changes to these procedures will be included in operator training to be conducted prior to implementation of the MELLLA+ amendment.

The NRC staff concludes that the normal licensee 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 EOPs and AOPs.

3.3.11 Plant-Specific MELLLA+ Dispositions Conclusion

The NRC staff concludes that, for the plant-specific dispositions discussed in SE Sections 3.3.2 through 3.3.10, the licensee has provided acceptable justification regarding operation in the MELLLA+ operating domain for PBAPS, Units 2 and 3.

3.4 Various Topics Based on RS-001

3.4.1 Introduction

As discussed above in SE Section 1.3, the NRC staff performed this review, in part, by using relevant sections of the review guidance in NRC, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001, Revision 0 (Reference 15). Although the MELLLA+ LAR is not an EPU, and RS-001 guidance is not wholly applicable, the NRC staff determined that RS-001 provides a good framework for the review of certain portions of the LAR.

RS-001 contains guidance for evaluating each area of review in the application, including the specific GDC used as the NRC's acceptance criteria. Since the guidance and SE template

contained in RS-001 are based on the final GDC and PBAPS, Units 2 and 3 were designed and constructed based on the draft GDC, Exelon submitted a supplement to the EPU application dated February 15, 2013 (Reference 34), which provided a revision to the RS-001 SE template. The revised SE template replaced references to the final GDC with the corresponding design criteria that constitute the current licensing basis for PBAPS. In preparing this MELLLA+ SE, the NRC staff used the SE template provided by the licensee for the EPU review.

The topics reviewed by the NRC staff, using RS-001 as guidance, are as follows:

- Fuel System Design (SE Section 3.4.2)
- Nuclear Design (SE Section 3.4.3)
- Thermal and Hydraulic Design (SE Section 3.4.4)
- Emergency Systems (SE Section 3.4.5)

Staff from Oak Ridge National Laboratory (ORNL) assisted the NRC staff in its review.

3.4.2 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for calculated performance; (2) final GDC 10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs; (3) draft GDC 37, 41, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided to prevent fuel damage following a LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The NRC staff has reviewed the impact on the fuel system of the proposed MELLLA+ operating domain expansion based on the licensee-provided analysis results for normal operation, AOOs, and infrequent and special events. The complete staff evaluation of these results is documented in SE Section 3.3.9, as supplemented by the information in SRXB-RAI-18 in Appendix A to this SE. As seen in that evaluation, operation at the lower MELLLA+ flows has an impact on transient response because the limiting event (i.e., LRNBP) occurs at the 83 percent core flow condition. To compensate for this increased AOO ΔCPR in the MELLLA+ region, the OLMCPR increase is comparable to 0.03. The licensee's analyses demonstrate that, with the proposed PBAPS MELLLA+ setpoints (i.e., increased OLMCPR), fuel damage is not expected for any AOO or the analyzed infrequent or special events, and core coolability will

be maintained. Thus, the NRC staff concludes that the impact on fuel of operation with the more restrictive setpoints at the lower MELLLA+ flows is minimal.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain expansion 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 expansion on the fuel system and demonstrated that (1) the fuel system is not likely to be damaged as a result of normal operation and AOOs; (2) the fuel system damage, should it happen, is not likely to be so severe as to prevent control rod insertion when it is required; (3) the number of fuel rod failures has not been underestimated for postulated accidents; and (4) coolability is likely to be maintained. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46; final GDC 10; and draft GDC 37, 41, and 44, following implementation of the proposed operating domain expansion. Therefore, the NRC staff finds the proposed operating domain expansion acceptable with respect to the fuel system design.

3.4.3 Nuclear Design

Regulatory Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and 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. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) final GDC 10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) draft GDC 8, insofar as it requires that the reactor core be designed so that the overall power coefficient in the power operating range shall not be positive; (3) final GDC 12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) draft GDC 12 and 13, insofar as they require that instrumentation and controls be provided as required to monitor and maintain variables within prescribed operating ranges through the core life; (5) draft GDC 14 and 15, insofar as they require that the protection system be designed to initiate the reactivity control systems automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and to initiate operation of ESFs under accident situations; (6) draft GDC 31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient, which could result in exceeding acceptable fuel damage limits; (7) draft GDC 27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (8) draft GDC 29 and 30, insofar as they require that at least one of the reactivity control systems be capable of making and holding the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (9) draft GDC 32, insofar as it requires that limits, which

include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the RCPB or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Operating Limits

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

The NRC staff reviewed the design changes between the PBAPS EPU core design and a reference MELLLA+ core design in terms of its impact on compliance with final GDC 10. The staff notes that the core and fuel design remain unchanged, and a full load of GNF2 fuel is used for both cores.

The SLMCPR is calculated based on the actual core loading pattern for each reload core; the results are reported in the SRLR (Reference 2). In the event that the cycle-specific SLMCPR is not bounded by the current PBAPS TS value, PBAPS must implement a license amendment to change the TS. As required by the M+ LTR, the SLMCPR is calculated at different operating conditions for every reload core. The specified conditions include: 100 percent CLTP and 100 percent Flow, 100 percent CLTP and 83 percent Flow, 100 percent CLTP and 110 percent Flow, and 78.8 percent CLTP and 55.0 percent Flow. The calculated SLMCPR values include the 0.02 adder required by the Methods LTR (Reference 10) for operation at power to flow ratios greater than 42 MWt/Mlbm/hr. The SLMCPR value for PBAPS will be evaluated for every reload, and it will continue to be evaluated on a cycle-specific basis prior to and after MELLLA+ implementation. For Cycle 21, which would be the first implementation, the SRLR specifies a SLMCPR value of 1.15 for both units. See SE Section 3.2.2 (SAR Section 2.2.1) for further discussion regarding the SLMCPR.

The OLMCPR is calculated by adding the change in MCPR (ΔCPR) due to the limiting AOO event to the SLMCPR. The OLMCPR for PBAPS is determined on a cycle-specific basis from the results of the reload transient analysis, which are documented in the SRLR. The final value of the OLMCPR is documented in the core operating limits report (COLR). Based on the generic results documented in the M+ LTR, and the reference transient analyses documented in Section 9 of the SAR, the MELLLA+ OLMCPR is comparable to 0.07 larger than an equivalent MELLLA core. The difference is due to (1) +0.03 in uncorrected ΔCPR/ICPR for the limiting AOO (i.e., LRNBP) occurring at the lower flow, (2) +0.02 for the Methods LTR SLMCPR penalty at powers greater than 42 MWt/Mlbm/hr, and (3) +0.02 for the SLO penalty on SLMCPR.

The LHGR and MAPLHGR operating limits are calculated for each reload fuel bundle design. The limits are documented in the cycle-specific COLR and on Tables 16.3-1 and 16.3-2 of the SRLR.

Section 4.3 of the SAR presents results for a large break LOCA analysis at different initial conditions. The evaluation concludes that MELLLA+ primarily affects the first peak in PCT, while the limiting peak is the second, which is controlled by decay heat and is minimally affected by operation in MELLLA+. Therefore, the limiting single failure is not affected by MELLLA+ operation. These LOCA analyses demonstrate that the limiting PCT values at MELLLA+ are within acceptable limits. The limiting single failure is not affected by MELLLA+ operation, and it remains the single battery failure. The evaluation concludes that PCT performance in PBAPS [[ ]] LOCA analyses are presented to demonstrate the PCT performance. The Appendix K second peak in PCT reaches [[ ]] this continues to meet the regulatory requirement. Mid-peak power shapes are the most limiting for large break LOCAs but still produce acceptable values.

Calculations were performed for a spectrum of break sizes for the small break LOCA at MELLLA+ conditions. [[ ]] The Cycle 21 SRLR concludes that the maximum Appendix K PCT is 1910 °F at EPU power and MELLLA+ core flow, while Appendix K PCT is 1905 °F at EPU power and rated core flow, [[ ]] The GNF2 Licensing Basis PCT is determined to be 1920 °F. [[ ]] The NRC staff finds that these results satisfy final GDC 10 and are therefore acceptable.

Monitoring and Control

Draft GDC 12 and 13 specify the requirements for instrumentation to monitor variables affecting the fission process. Maneuvering within the MELLLA+ operating domain is performed by either controlling the recirculation flow or moving control rods. GDC 13 requires that instrumentation be provided to ensure that the operation is within prescribed operating ranges.

The design changes to incorporate MELLLA+ do not include any changes to the neutron monitoring system (NMS) or the flow instrumentation. Nevertheless, the staff reviewed the effects of operation in the expanded domain on instrumentation performance and the adequacy of the NMS to meet the requirements of draft GDC 12 and 13.

At the MELLLA+ low corner (Point K of Figure 3.1-1), the power-to-flow ratio is maximized and there is the potential to encounter void formation in the bypass region. The NRC staff requested that the licensee determine the effect of bypass void formation on local power range monitors (LPRMs) and traversing incore probes (TIPs). The licensee's evaluation concluded that no bypass voiding is expected at the highest TIP elevation that corresponds to the highest LPRM detector elevation. This value is calculated using the ISCOR hot channel methodology, which is conservative, because it neglects cross flow between bundles in the bypass region.

Based on the above, the NRC staff finds that the PBAPS instrumentation and control systems are adequate to fulfill the requirements of draft GDC 12 and 13 under MELLLA+ conditions.

Reactivity Control

Draft GDC 14, 15, 27, 28, 29, 30, 31, and 32 specify the requirements for the reactivity control systems.

Power control is achieved in the MELLLA+ expanded operating domain by controlling core reactivity with control blades, as well as recirculation flow.

Draft GDC 14, 15, and 31 are met by the reactor protection system and the scram function of the control rod system. These are unaffected by the implementation of the MELLLA+ domain. Therefore, the NRC staf finds that draft GDC 14, 15, and 31 are acceptably met.

Draft GDC 27, 28, 29, and 30 are met by the control rod drive system and the SLC system. These systems are unaffected by the implementation of the MELLLA+ domain. Therefore, the NRC staff finds that draft GDC 27, 28, 29, and 30 are acceptably met.

Compliance with draft GDC 32 is assured by demonstrating acceptable radiological consequences and barrier integrity during postulated control rod drop accidents. The most limiting conditions occur during low power operation and are, therefore, unaffected by the MELLLA+ implementation. Therefore, the NRC staff finds that GDC 32 is acceptably met.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed operating domain expansion 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 expansion 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 applicable requirements of final GDC 10 and 12 and draft GDC 8, 12, 13, 14, 15, 27, 28, 29, 30, 31, and 32. Therefore, the NRC staff finds the proposed operating domain expansion acceptable with respect to the nuclear design.

3.4.4 Thermal and Hydraulic Design

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, which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and DBA conditions and core thermal-hydraulic stability under normal operation and ATWS events. The NRC's acceptance criteria are based on (1) final GDC 10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded

during any condition of normal operation, including the effects of AOOs, and (2) final GDC 12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Analytical Methods

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

1. The ISCOR code does not have an explicitly approved SE; however, the SE for NEDE-24011-P/Revision 0, mentions a "digital computer code" that is an acceptable method. GEH confirmed that the digital computer code referred to in the NEDE-24011-P/Revision 0 SE is indeed ISCOR.
2. A similar situation occurs with the STEMP code, which is used to calculate the suppression pool temperature using basic energy conservation equations. STEMP was referenced in the approval of NEDE-24222.
3. The LAMB code is explicitly approved for use in ECCS-LOCA applications, but it is not explicitly approved for use in reactor internal pressure differences and containment response. However, this is simply an extension of the approved use, and the models used are the same as those of the approved ECCS-LOCA application.
4. TRACG04 is currently approved for use in DSS-CD and ATWS analysis and has been used for ATWSI best estimate calculations; however, the licensing basis ATWS long-term analyses are based on ODYN.
5. Following approval of Amendment 26 of GESTAR II, GEH implemented TGBLA06 and PANAC11.

Thus, the NRC staff concludes that all the methods used in the SAR are either approved or an acceptable extension of an approved code.

Equivalency to Proven Designs

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

Steady State Operation

The following Tables 3.4.4-1 and 3.4.4-2 show a summary of the PBAPS steady state operating conditions (extracted from Tables 1-2 and 1-3 of the SAR). The volumetric power density in PBAPS is 58.45 kW/L.

Table 3.4.4-1 - PBAPS Operating Conditions

Parameter	**MELLLA 100% CLTP, 99% CF**	**MELLLA+ 100% CLTP, 83% CF**	**MELLLA+ 78.8% CLTP, 55% CF**
Thermal Power (MWt)	3,951	3,951	3,113
Dome Pressure (psia)	1,050	1,050	1,017
Steam Flow Rate (Mlb/hr)	16.169	16.150	12.347
FW Flow Rate (Mlbm/hr)	16.137	16.118	12.315
FW Temperature (°F)	381.5	381.4	359.5
Core Flow (Mlbm/hr)	101.475	85.075	56.375
Core Inlet Enthalpy (BTU/lbm)	521.2	515.1	500.1
Core Pressure Drop (psi)	24.4	18.8	13.4
Core Average Void Fraction	0.48	0.51	0.51
Average Core Exit Void Fraction	0.71	0.74	0.76

Table 3.4.4-2 - PBAPS Power-to-Flow Ratios

Operating Domain	**Point on Power/Flow Map**	**Core Thermal Power (MWt/%CLTP)**	**CF (Mlbm/hr/ % rated CF)**	**Power to Flow Ratio (MWt/Mlbm/hr)**
Current Operating Domain 100% RCF	E	3951 / 100%	102.5 / 100%	38.5
Current Operating Domain 99% RCF	D	3951 / 100%	101.5 / 99%	38.9
MELLLA+ Operating Domain 83% RCF	J	3951 / 100%	85.1 / 83%	46.4
MELLLA+ Operating Domain 55% RCF	K	3113 / 78.8%	56.4 / 55%	55.2

As shown in Table 3.4.4-1, the power density at Point K (78.8 percent power and 55 percent flow) is 55.2 MWt/Mlbm/hr, which is greater than the action threshold of 50 MWt/Mlbm/hr set by Limitation 9.3 of the Methods LTR. During the evaluation of the Methods LTR, the NRC staff reviewed power distribution uncertainties up to power-flow ratios of 44 MWt/Mlbm/hr and gamma scan results. The staff found that an uncertainty of 0.02 should be added to the SLMCPR to cover operation above 42 MWt/Mlbm/hr (see Figure 3.4.4-1 below). A SLMCPR uncertainty of at least 0.01 must be applied below 42 MWt/Mlbm/hr, based on Limitation 9.5 of the Methods LTR. The reason for this limitation is that insufficient data was available to judge power distribution uncertainties at the higher void fraction levels. Extremely high void fractions result in increased errors in cross-section generation and challenge some of the assumptions in modern nodal neutronic methods because of the harder neutron spectrum. PBAPS has chosen to conservatively apply a 0.02 SLMCPR uncertainty in all areas of the power-flow map.

Figure 3.4.4-1 - Illustration of Power-to-Flow Ratio Requirements from Limitation 9.3

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Table 2-2 of the PBAPS SAR provides the ISCOR calculation results that calculate no bypass voiding at the Instrumentation D Level for 100 percent CF, 99 percent CF, and 83 percent CF at 100 percent CLTP (points E, D, and J, respectively).

For operation at power-flow greater than 50 (for example, at point K in Figure 3.4.4-1), the NRC staff SE required a case-specific evaluation to ensure that the particular plant does not have

unusual uncertainty values. The staff also required an additional penalty on the SLMCPR by using SLO uncertainties, even though SLO operation is not allowed under MELLLA+ conditions. This restriction applies to the entire MELLLA+ region in Figure 3.4.4-1 (the region defined by points D, J, K, and L), but points D and L lie on the non-MELLLA+ region. Therefore, the SLO uncertainty is only required at points K and J. PBAPS has conservatively applied SLO uncertainties to all the points in the power-flow map.

Following the guidelines from Limitation 9.3 of the Methods LTR, the NRC staff has reviewed, on a plant-specific basis, the power distribution uncertainties for PBAPS. This review was based on a comparison of TIP data provided by the licensee against PANACEA (core simulator computer code) calculated power distributions. The TIP data encompasses five cycles from 2004 to 2014. Figure 3.4.4-1 shows the locations in the power-flow map where TIP measurements were performed in PBAPS for the last five cycles (i.e., Cycles 16-20). Analysis of this data show that the power distribution root mean squared (RMS) error is between 2.5 percent for radial power and 8 percent for nodal power. The detailed TIP data was provided in the licensee's letter dated April 28, 2015 (Reference 4).

Transient Response

The licensee has provided analyses for normal operation, AOOs, and special events. The complete NRC staff evaluation of these results is documented SE Section 3.3.9. As seen in that evaluation, operation at lower flows in the MELLLA+ domain has a small impact (~0.03 in ΔCPR/ICPR) on transient response, and the limiting initiating conditions are at 83 percent core flow for all AOOs analyzed. Calculations show that the limiting AOO is the LRNBP (see Table 9-1 of the SAR). For this case, the ΔCPR/ICPR is 0.16 when initiating from the 110 percent flow conditions and 0.19 when initiating from 83 percent flow condition, which implies a potential +0.03 penalty increase in uncorrected ΔCPR/ICPR for operation in the MELLLA+ domain.

Stability

PBAPS will implement the DSS-CD solution consistent with the M+ LTR. DSS-CD implementation includes any limitations and conditions in the applicable DSS-CD LTR. PBAPS has a full-core load of GNF2 fuel; therefore, the transition from Option III to DSS-CD does not require any special analyses because they are covered by the DSS-CD LTR results.

[[

]] The conservative calculations documented in Table 2-6 of the SAR confirm that these [[ ]] values are acceptable because they follow the approved procedure established in the DSS-CD LTR, and the calculated final MCPR is greater than the SLMCPR [[

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In the SRLR, the licensee provided the preliminary PBAPS backup stability protection (BSP) regions, which are calculated using approved procedures.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain expansion on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed operating domain expansion on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed operating domain expansion on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of final GDC 10 and 12, following implementation of the proposed operating domain expansion. Therefore, the NRC staff finds the proposed operating domain expansion acceptable with respect to thermal and hydraulic design.

3.4.5 Emergency Systems

The following topics were evaluated as part of the emergency systems review:

- Functional Design of the Control Rod Drive System (SE Section 3.4.5.1)
- Overpressure Protection during Power Operation (SE Section 3.4.5.2
- Reactor Core Isolation Cooling System (SE Section 3.4.5.3)
- Residual Heat Removal System (SE Section 3.4.5.4)
- Standby Liquid Control System (SE Section 3.4.5.5)

3.4.5.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the control rod drive (CRD) system to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRD system cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on the following: (1) Draft GDC 40 and 42, insofar as they require that protection be provided for engineered safety features (ESFs) against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA, (2) Draft GDC 26, insofar as it requires that the protection system be designed to fail into a safe state, (3) Draft GDC 31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient, which could result in exceeding acceptable fuel damage limits, (4) Draft GDC 27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits, (5) Draft GDC 29 and 30, insofar as they require that at least one of the reactivity control systems be capable of making and holding the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits, (6) Draft GDC 32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods

or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (7) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system and that the ARI system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

As discussed in SAR Section 2.5, no change is made to the control rods or CRD system due to the proposed MELLLA+ operating domain expansion. In addition, the CRD system will not be negatively impacted by operation in the MELLLA+ operating domain. Therefore, the regulatory requirements in draft GDC 26, 27, 28, 29, 30, 31, 32, 40, and 42 and 10 CFR 50.62(c)(3) continue to be satisfied by the design.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain expansion on the functional design of the CRD system. The NRC staff concludes that the CRD system safety functions will not be impacted by MELLLA+ operation. As such, the regulatory requirements in draft GDC 26, 27, 28, 29, 30, 31, 32, 40, and 42 and 10 CFR 50.62(c)(3) will continue to be satisfied by the design.

3.4.5.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covered relief and safety valves on the main steam lines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) draft GDC 9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime, and (2) final GDC 31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2.

Technical Evaluation

The licensee has evaluated the impact of the proposed operating domain expansion on overpressure protection. The evaluation is documented in Section 3.1 of the SAR. For PBAPS, the limiting overpressure event is main steam isolation valve closure followed by a high-flux scram. Analyses in Section 3.1 of the SAR indicate that the peak RPV pressure remains essentially unchanged. The 3 percent SRV tolerance assumed in the PBAPS ASME overpressure event analysis is unchanged from pre-MELLLA+ assumptions, and it is consistent with the actual SRV performance testing at PBAPS. The resulting peak dome pressure is 1,324 psig, and the peak RPV pressure is 1,352 psig. This peak RPV pressure is below the ASME Service Level B limit of 1,375 psig. Based on this, the NRC staff concludes that the

overpressure protection features will continue to meet final GDC 31 and draft GDC 9, following implementation of the proposed LAR. Therefore, the NRC staff finds the proposed LAR acceptable with respect to overpressure protection during power operation.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain expansion on the overpressure protection capability of the plant during power operation. The NRC staff concludes that the licensee has (1) adequately accounted for the effects of the proposed operating domain expansion on pressurization events and overpressure protection features and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the overpressure protection features will continue to meet final GDC 31 and draft GDC 9, following implementation of the proposed operating domain expansion. Therefore, the NRC staff finds the proposed operating domain expansion acceptable with respect to overpressure protection during power operation.

3.4.5.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The reactor core isolation cooling (RCIC) system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main 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 (SBO). The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covered the effect of the proposed MELLLA+ operating domain expansion on the functional capability of the system. The NRC's acceptance criteria are based on (1) draft GDC 40, insofar as it requires that protection be provided for ESFs against dynamic effects; (2) draft GDC 4, insofar as reactor facilities shall not share systems or components unless it is shown that safety is not impaired by the sharing; (3) draft GDC 37, insofar as it requires that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems; (4) draft GDC 51, and 57, insofar as they require that piping systems penetrating containment be designed with appropriate features as necessary to protect from an accidental rupture outside containment and the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (5) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from an SBO of a specified duration. Specific review criteria are contained in SRP Section 5.4.6.

Technical Evaluation

As discussed in SAR Section 3.9, there are no changes to the RCIC system design as a result of the proposed MELLLA+ operating domain expansion. In addition, there are no changes to the normal reactor operating pressure, decay heat, or SRV setpoints as a result of the proposed MELLLA+ operating domain expansion. With respect to NPSH, there are no physical changes to the RCIC pump suction configuration as a result of the proposed MELLLA+ operating domain expansion. There are also no changes to the RCIC flow rate or minimum atmospheric pressure in the suppression chamber and the condensate storage tank. Based on these considerations,

the NRC staff concludes that the functional capability of the RCIC system is not impacted by the LAR.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain expansion on the ability of the RCIC system to provide decay heat removal following an isolation of main FW event and a station blackout event and the ability of the system to provide makeup to the core following a small break in the RCPB. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed operating domain expansion on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed operating domain expansion. Based on this, the NRC staff concludes that the RCIC system will continue to meet the requirements of draft GDC 4, 37, 40, 51, and 57 and 10 CFR 50.63, following implementation of the proposed operating domain expansion. Therefore, the NRC staff finds the proposed operating domain expansion acceptable with respect to the RCIC system.

3.4.5.4 Residual Heat Removal System

Regulatory Evaluation

The residual heat removal (RHR) system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system that takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covered the effect of the proposed MELLLA+ operating domain expansion on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) draft GDC 40 and 42, insofar as they require that protection be provided for ESFs against dynamic effects, and (2) draft GDC 4, insofar as reactor facilities shall not share systems or components unless it is shown that safety is not impaired by the sharing. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The RHR system design is not impacted by the proposed MELLLA+ operating domain expansion. In addition, there are no changes to decay heat as a result of the proposed MELLLA+ operating domain expansion. The RHR pumps have been analyzed for plant-specific conditions with uncertainties documented in Section 4.2.6.1 of the SAR and have sufficient NPSH margin to perform satisfactorily during containment isolation events initiated at MELLLA+ conditions. Thus, the requirements of draft GDC 4, 40, and 42 will continue to be satisfied.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain expansion on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed operating domain expansion on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the

RHR system will continue to meet the requirements of draft GDC 4, 40, and 42, following implementation of the proposed operating domain expansion. Therefore, the NRC staff finds the proposed operating domain expansion acceptable with respect to the RHR system.

3.4.5.5 Standby Liquid Control System

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 effect shutdown. The NRC staff's review covered the effect of the proposed MELLLA+ operating domain expansion on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) draft GDC 27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (2) draft GDC 29 and 30, insofar as they require that at least one of the reactivity control systems be capable of making and holding the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (3) 10 CFR 50.62(c)(4), insofar as it requires that the SLC system be capable of reliably injecting a borated water solution into the RPV at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The hot shutdown boron weight (HSBW) is calculated on a generic basis for each fuel line (e.g., GNF2 in the case of PBAPS). The HSBW is confirmed effective on plant- and cycle-specific bases with ODYN and TRACG ATWS calculations. Section 9.3.1 of the SAR documents these calculations. Both the licensing bases and the best estimate ATWS calculations show that the generic HSBW is effective to shut down the PBAPS core under MELLLA+ initial conditions.

The Boron-10 enrichment of the SLC system has been increased from 62 atom percent to 92 atom percent as part of the EPU. With this change, the time to inject the HSBW and place the reactor in hot shutdown has been decreased significantly (reducing the time by about a third). The enrichment change positively enhanced the safety of PBAPS.

The thermal-hydraulic parameters (flow, pressure, temperature) of the SLC system design have not been modified relative to the baseline, the reactor pressure has not been modified, and the SLC system boron inventory shutdown margin has been evaluated for the initial core in the SAR. Therefore, the NRC staff finds that the requirements of draft GDC 27, 28, 29, and 30 and 10 CFR 50.62(c)(4) will continue to be satisfied relative to SLC system operation.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain expansion on the SLC system and concludes that the licensee has adequately accounted for the effects of the proposed operating domain expansion 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 expansion. Based on this, the NRC staff concludes that the SLC system will continue to meet the requirements of draft GDC 27, 28, 29, and 30 and 10 CFR 50.62(c)(4), following implementation of the proposed operating domain expansion. Therefore, the NRC staff finds the proposed operating domain expansion acceptable with respect to the SLC system.

3.5 Limitations and Conditions from Applicable LTRs

SAR Appendices A, B, C, and D summarize the licensee's disposition of the limitations and conditions in the NRC staff SEs for the following LTRs:

Appendix A: NEDC-33173P-A (Methods LTR)

Appendix B: NEDC-33006P-A (M+ LTR)

Appendix C: NEDC-33075P-A (DSS-CD LTR)

Appendix D: NEDE-32906P-A, Supplement 1-A (TRACG for ATWS Overpressure)

SE Sections 3.5.1 through 3.5.4 provide a brief description of the licensee's disposition for the limitations and conditions discussed in SAR Appendices A, B, C, and D. The NRC staff conclusion regarding the limitations and conditions is provided in SE Section 3.5.5.

3.5.1 Methods LTR - NEDC-33173P-A Limitations and Conditions

Appendix A of the SAR summarizes the licensee's disposition of the limitations and conditions in the NRC staff SE for the Methods LTR, NEDC-33173P-A (Reference 10).

The following limitations and conditions do not apply to PBAPS:

- 9.2, 3D Monicore - This limitation is only applicable to an application relying on TGBLA04/PANAC10. The PBAPS MELLLA+ application is based on TGBLA06 and PANAC11.

- 9.4, SLMCPR 1 - This limitation has been superseded by Revision 4 of the Methods LTR. Revision 4 requires that a 0.01 value shall be added to the cycle-specific SLMCPR value for power-to-flow ratios up to 42 MWt/Mlbm/hr, and a 0.02 value shall be added to the cycle-specific SLMCPR value for power-to-flow ratios above 42 MWt/Mlbm/hr. PBAPS has chosen to conservatively add a 0.02 value to the cycle-specific SLMCPR above and below 42 MWt/Mlbm/hr.

- 9.13, Application of 10 Weight Percent Gadolinium – This limitation is not applicable because PBAPS uses GNF2 fuel, and as such, does not seek to apply 10 weight percent Gadolinium to this licensing application.

- 9.14, Part 21 Evaluation of GESTR-M Fuel Temperature Calculation – This limitation is not applicable because the 10 CFR Part 21 report referred to herein is related to

GESTR-M thermal-mechanical (T-M) evaluation, while the PBAPS evaluation is based on GNF2 fuel, which has a PRIME T-M model.

- 9.16, Void Reactivity 2 – This limitation is not applicable because the SAR uses void reactivity coefficients bias and uncertainties from the latest version of TRACG for the GNF2 lattice designs loaded in the core.

- 9.18, Stability Setpoints Adjustment – This limitation is not applicable. The 5 percent OPRM and 2 percent APRM calibration errors caused by the possible presence of bypass voids were taken into account in the DSS-CD LTR and [[ ]] approved in the DSS-CD LTR SE.

- 9.19, Void-Quality Correlation 1 – This limitation is not applicable because the most recent TRACG AOO and ATWS Overpressure Methodology used, and the related SE removed, the 0.01 penalty for TRACG applications.

- 9.21, Mixed Core Method 1 – This limitation is not applicable because the LAR is not implementing MELLLA+ with a mixed fuel vendor core.

- 9.22, Mixed Core Method 2 – This limitation is not applicable because GNF2 is an approved fuel line in the Methods LTR.

- 9.23, MELLLA+ Eigenvalue Tracking – This requirement was clarified in a GEH letter dated August 26, 2015 (Reference 35). GEH stated that the limitation and condition is a GEH requirement to gather and evaluate data. It is not a plant-specific requirement. As such, this requirement is not applicable to PBAPS.

The following limitations and conditions apply to PBAPS:

- 9.1, TGBLA/PANAC Version – This limitation is met by the use of TGBLA06/PANAC11 methods.

- 9.3, Power-to-Flow Ratio – The PBAPS power density can be as high as 55.2 MW/Mlbm/hr in the MELLLA+ operating domain, which exceeds the 50 MW/Mlbm/hr action threshold. This limitation is addressed by (1) applying an additional uncertainty to the SLMCPR calculation using the SLO flow uncertainties, which is equivalent to applying a +0.02 penalty, and (2) evaluating plant-specific power distribution uncertainties based on PBAPS TIP measurements.

- 9.5, SLMCPR 2 – A 0.02 penalty is added to the PBAPS SLMCPR for operation above 42 MWt/Mlbm/hr in accordance with the Methods LTR. A 0.01 penalty would be applicable below 42 MWt/Mlbm/hr, but PBAPS has conservatively applied the 0.02 penalty in the complete power-flow map.

- 9.6, R-Factor – The R-factor calculation is consistent with the axial void profiles expected in PBAPS.

- 9.7, ECCS-LOCA 1 – The PBAPS ECCS LOCA analyses include an evaluation for top-peaked and mid-peaked axial power profiles.

- 9.8, ECCS-LOCA 2 – PBAPS ECCS LOCA calculations have been performed at the required statepoints, including the MELLLA+ corners, and have demonstrated compliance with limits.

- 9.9, Transient LHGR 1 –The plant-specific analysis demonstrated that T-M limits are satisfied with the proposed operating margins for the MELLLA+ core.

- 9.10, Transient LHGR 2 – The SRLR was submitted to the NRC staff for review, and the T-M response was included in Appendix B of the SRLR. T-M limits are satisfied with the proposed operating margins for the MELLLA+ core.

- 9.11, Transient LHGR 3 – Since the void history bias is incorporated into the transient model within TRACG, the additional 10 percent bias to the fuel centerline melt and the 1 percent cladding circumferential plastic strain is no longer required.

- 9.12, LHGR and Exposure Qualification – The PBAPS MELLLA+ analysis utilizes PRIME T-M methods.

- 9.15, Void Reactivity 1 – Void reactivity coefficient bias and uncertainties used in the latest version of TRACG are applicable to the GNF2 lattice designs loaded in the core.

- 9.17, Steady-State 5 Percent Bypass Voiding – Bypass voiding was evaluated and calculated to be below the 5 percent limit at all LPRM levels, which satisfies the limitation.

- 9.20, Void-Quality Correlation 2 – The TRACG interfacial shear model complies with the NRC SE for NEDE-32906P, Supplement 3-A (Reference 13), as required by this limitation.

- 9.24, Plant-Specific Application – The bundle power, operating LHGR, and MCPR have been provided for the equilibrium GNF2 MELLLA+ PBAPS cycle. All limits are satisfied.

3.5.2 M+ LTR - NEDC-33006P-A Limitations and Conditions

Appendix B of the SAR summarizes the licensee's disposition of the limitations and conditions in the NRC staff SE for the M+ LTR, NEDC-33006P-A (Reference 7).

The following limitations and conditions do not apply to PBAPS:

- 12.18.a through f, ATWS TRACG Analysis - Best-estimate ATWS with depressurization is not required because (1) the ODYN licensing calculation does not reach the HCTL limit, so depressurization is not required, and (2) PBAPS has increased the B-10 enrichment to 92 atom percent.

- 12.20, Generic ATWS Instability – This limitation is not applicable because PBAPS has performed a plant-specific ATWSI evaluation and does not use the generic ATWSI analyses.

The following limitations and conditions apply to PBAPS:

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

- 12.2, Related LTRs - The limitations from NEDC-33173P-A, NEDC-33075P-A, and NEDC-32906P, Supplement 3-A, are specifically addressed in Appendices A, C, and D of the SAR. Limitations of NEDC-33147P-A are no longer addressed since TRACG is now approved for DSS-CD stability solution calculations.

- 12.3, Concurrent Changes

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

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

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

    - 12.3d – ATWS instability analyses supporting the M+ condition are based on the GNF2 fuel response.

    - 12.3e – GNF2 was approved for expanded operating domain in Supplement-3 of the Methods LTR, NEDC-33173P-A and new analyses were performed for a specific core configuration.

    - 12.3f - PBAPS will have a full load of GNF2 fuel. The DSS-CD resolution has been updated with GNF2 analysis. Conditions have been met.

    - 12.3g – DSS-CD will be employed in PBAPS to address possible instabilities. DSS-CD has been approved for MELLLA+ applications with GNF2 fuel.

- 12.4, Reload Analysis Submittal – The PBAPS application provided the plant-specific thermal limits and transient assessment in the SRLR.

- 12.5, Operating Flexibility

    - 12.5a – SLO operation is not allowed in the MELLLA+ operating domain. The PBAPS TSs will be revised accordingly as part of the MELLLA+ implementation.

    - 12.5b – FWHOOS will be allowed in PBAPS under MELLLA+ conditions. A license condition of operating within ±10 °F of the nominal FW temperature will be implemented in PBAPS as part of the MELLLA+ implementation. The FWHOOS restriction in the M+ LTR SE was imposed to minimize stability concerns at the low-flow (55 percent) statepoint. TRACG analyses have been performed at nominal FW and off-rated at -10 °F, which is the maximum allowed

FW temperature reduction in MELLLA+. In addition, a -90 °F FW temperature case was evaluated from initial conditions outside MELLLA+. The analyses demonstrate that DSS-CD provides sufficient level of protection in case of instabilities for both FW temperature bounds.

  - 12.5c – The licensee has provided the power-flow map in the SAR, and it will be included in the COLR.

- 12.6, SLMCPR Statepoints and CF Uncertainty – The licensee has evaluated the SLMCPR at off-nominal conditions, including the 55 percent flow statepoint, and has reported it in the SRLR.

- 12.7, Stability – The DSS-CD automated BSP option will be implemented at PBAPS. If the primary OPRM DSS-CD scram is declared inoperable, manual BSP will be implemented immediately by reducing the operating power line until the automated BSP option can be activated in the DSS-CD computer.

- 12.8, Fluence Methodology and Fracture Toughness – The change of vessel effective full power years (EFPY) is estimated to be [[ ]] at 54 EFPY under MELLLA+ conditions [[ ]]. An up-to-date approved methodology was used for the estimation.

- 12.9, RCPB – A discussion of non-category-A materials and adequacy of the augmented inspection program is presented in Section 3.5.1.4 of the SAR.

- 12.10, ECCS-LOCA Off-Rated Multiplier

  - 12.10a, b, c, and d – PBAPS-specific Appendix K ECCS LOCA calculations were provided in the SAR. PBAPS is small-break LOCA limited. Both Appendix K and nominal PCT values are reported, and the Appendix K values bound the results. [[ ]]

- 12.11, ECCS-LOCA Axial Power Distribution Evaluation – Top peaked and mid-peaked power shapes have been used for the LOCA analyses.

- 12.12, ECCS LOCA Reporting

  - 12.12a and b – Both the nominal and the Appendix K LOCA results have been reported in the SAR.

- 12.13 and 12.14, Small Break LOCA and Break Spectrum – PBAPS is small break LOCA limited at the MELLLA+ minimum core flow conditions, which has been analyzed and reported.

- 12.15, Bypass Voiding above the D-Level – No bypass voiding is expected at either the top of the TIP instrument or the D-level LPRM.

- 12.16, Rod Withdrawal Error – A plant-specific rod withdrawal error analysis was performed using PANACEA to confirm the validity of the rod block monitor setpoints.

- 12.17, ATWS LOOP – ATWS calculations were performed in Section 9.3.1 of the SAR using the licensing basis (ODYN). TRACG best estimate calculations are not required because the HCTL limit is not reached and the B-10 enrichment was increased to 92 atom percent. LOOP analysis is not required because RHR is not affected by LOOP.

- 12.19, Plant-Specific ATWS Instability – The licensee has provided a best estimate ATWSI calculation using TRACG04 to demonstrate compliance with limits.

- 12.21, Individual Plant Evaluation – A plant-specific probabilistic risk assessment was included in Section 10.5 of the SAR.

- 12.22, Irradiated Assisted Stress Corrosion Cracking – Plant-specific fluence calculation results provided in Section 10.7 of the SAR indicate that the top guide, core plate, and shroud exceed the 5E20 n/cm^2 threshold for irradiated assisted stress corrosion cracking (IASCC). However, the current inspection strategies in place are considered sufficient to address IASCC of reactor internals.

- 12.23 Limitations from the ATWS RAI Evaluations

  - 12.23.1 – Best estimate ATWS depressurization analyses with TRACG04 are not required (see Limitation 12.18d).

  - 12.23.2 – The ATWS calculations key parameters were provided.

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

  - 12.23.4 – The EOP procedures were reviewed, and sensitivity analyses were performed, for different water level control strategies. The EOPs require the operator to lower level to the top of active fuel (unless the transient terminates early) and control within a band between the minimum steam cooling water level and 2 feet below the spargers. A wide band is necessary because manual level control during an ATWS cannot be accomplished accurately. The sensitivity calculations indicate that the EOP strategy is adequate to satisfy the ATWS criteria.

  - 12.23.5 – The PBAPS power density at the full power-minimum flow statepoint in the MELLLA+ operating domain (Point J of Figure 3.1-1) is 46.4 MWt/Mlbm/hr, which does not exceed the 52.5 MWt/Mlbm/hr limit.

  - 12.23.6 – ATWS instability analysis was performed for GNF2 fuel as required by the MELLLA+ LTR SE for fuels other than GE14.

  - 12.23.7 – ATWS instability analysis was performed for GNF2 fuel because PBAPS has a full load of GNF2 fuel.

  - 12.23.8 – The ATWS calculations accounted for all PBAPS-specific features.

- 12.23.9 – The plant-specific ATWS calculations accounted for the physical limitations of ECCS systems used (HPCI and RCIC for PBAPS).

- 12.23.10 – The containment pressure calculated by the licensing basis ODYN analysis is [[ ]] given in SAR Table 9-3, which is under the containment limit of 56 psig for PBAPS. All safety grade equipment will function under these containment pressure conditions.

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

- 12.24, Limitations from Fuel Dependent Analyses RAI Evaluations

  - 12.24.1 – The TRACG PBAPS-specific calculations model the water rod flow explicitly.

  - 12.24.2 – The average core exit void fraction is presented in Table 1-2 of the SAR for a MELLLA and MELLLA+. The highest void fraction under MELLLA+ corresponds to the low flow point (78.8 percent CLTP, 55 percent Flow) and has a value of 76 percent, compared to 71 percent for the nominal MELLLA condition (100 percent CLTP, 99.0 percent Flow).

  - 12.24.3 - The licensee has evaluated the SLMCPR at off-nominal conditions, including the 55 percent flow statepoint and has reported it in the SRLR. The value is 1.15 for both units.

  - 12.24.4 – Best-estimate ATWS with depressurization is not required in PBAPS because HCTL limit is not reached and the B-10 enrichment has been increased to 92 atom percent.

3.5.3 DSS-CD LTR - NEDC-33075P-A Limitations and Conditions

Appendix C of the SAR summarizes the licensee's disposition of the limitations and conditions in the NRC staff SE for the DSS-CD LTR, NEDC-33075P-A (Reference 8) as follows:

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

- 5.2 – The DSS-CD CDA setpoint calculation followed the procedure outlined in the DSS-CD LTR.

- 5.3 – The values of the FIXED and ADJUSTABLE parameters are established by GEH and will be documented in a DSS-CD Settings Report.

- 5.4 – Verification and validation (V&V) of the DSS-CD trip function code was performed for transportability considerations.

It must be noted that the previous version of DSS-CD LTR, NEDC-33075P-A, Revision 6, relied on a separate LTR, NEDE-33147P-A, for the TRACG application. When Revision 7 of the DSS-

CD LTR was issued, it incorporated the TRACG application, and the limitations of that LTR are no longer applicable.

3.5.4 TRACG Application for ATWS Overpressure LTR - NEDE-32906P-A, Supplement 1-A

Appendix D of the SAR summarizes the licensee's disposition of the limitations and conditions associated with the application to the use of TRACG04/PANAC11 in the analysis of the ATWS overpressure event. The four limitations and conditions from the NRC SE of LTR NEDE-32906P-A, Supplement 1-A (Reference 12), are addressed in Table D-1 as follows:

- 4.1, Vessel peak pressure only - The methodology is only used for peak pressure at the vessel location before the time of SLC system initiation.

- 4.2, Range of applicability: power/flow - The actual power and flow used in the calculations are within the approved LTR ranges. Since PBAPS is not a BWR/2, the range of applicability is satisfied.

- 4.3, Power in absolute units - The power used was 3,951 MWt, which corresponds to 100 percent CLTP and 120 percent OLTP.

- 4.4, No Instability - The overpressure conditions are evaluated early into the transient before unstable power oscillations have time to develop.

3.5.5 Limitations and Conditions Conclusion

The NRC staff has reviewed the licensee's disposition of the limitations and conditions discussed in Appendices A, B, C, and D of the SAR as discussed in SE Section 3.5.1 through 3.5.4. The staff concludes that the licensee has met the applicable limitations and conditions.

3.6 Use of TRACG

The NRC staff has reviewed the TRACG code models and concludes that TRACG calculations of ATWSI for PBAPS with possible rewetting and quenching is sufficient to provide reasonable assurance of compliance with the applicable ATWS regulatory criteria – namely, demonstrating that core coolability is maintained during ATWSI events. This staff review considered plant-specific information (e.g., EOPs), specific aspects of the TRACG code use as it was applied in the context of the PBAPS ATWSI analysis provided by the licensee (e.g., updates to the quench model, revision to the Tmin correlation in TRACG, etc.) and justification of the applicability of experimental data. The current review does not constitute a generic review and approval of the TRACG method.

While the NRC staff questioned the licensee, in SRXB-RAI-18, about the acceptability of the Modified Shumway Tmin correlation in TRACG, the licensee demonstrated acceptable results with the conservative Homogeneous Nucleation Temperature Tmin correlation and realistic reactor and plant system transient response characteristics. Therefore, the use of TRACG for the PBAPS ATWSI calculation is acceptable.

4.0 LICENSE AND TECHNICAL SPECIFICATION CHANGES

The following provides the NRC staff evaluation of the proposed RFOL and TS changes associated with the LAR. The proposed changes are shown in Attachment 2 to the application dated September 4, 2014, as supplemented by the licensee's letters dated October 26, 2015, and January 15, 2016.

4.1 License Condition - Feedwater Temperature

The RFOL would be revised to add new License Condition 2.C(16), which would read as follows:

> (16) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration
>
> The licensee shall not operate the facility within the MELLLA+ operating domain with a feedwater heater out of service resulting in more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

The licensee has proposed this license condition to satisfy Limitation and Condition 12.5.b in the NRC staff's SE for the M+ LTR. This limitation and condition states:

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

As discussed in SAR Section 2.4.4, the intent of the restriction is to limit the core inlet subcooling and prevent degradation of the stability response in the event of a recirculation pump trip. The licensee stated that the license condition would prohibit the facility from operating with a feedwater heating capacity less than that required to produce a FW temperature of 371.5 °F. As discussed on page 26 of Attachment 1 to the licensee's letter dated April 28, 2015, [[

]]

The NRC staff concludes that the proposed license condition is acceptable since it meets the intent of M+ LTR Limitation and Condition 12.5.b and is supported by the licensee's MELLLA+ analysis. See Appendix A, SRXB-RAI-3, for further discussion regarding the licensee's analysis related to this FW temperature limitation.

4.2 Technical Specification Changes

The licensee submitted changes to the PBAPS TSs to support the MELLLA+ LAR as shown in Attachment 2 to the application dated September 4, 2014. The proposed TS changes are primarily associated with implementation of the DSS-CD stability solution. The changes associated with implementation of the DSS-CD stability solution are discussed in Section 8.0,

"Effect of Technical Specifications," of the DSS-CD LTR (Reference 8). Example TSs are shown in Appendix A (redline/strikeout version) and Appendix B (revision bar version) of the LTR. Section 3.2.5 of the NRC staff SE for the DSS-CD LTR found the example TS changes to be acceptable based on the technical content. However, the staff's SE stated that the example TSs in the DSS-CD LTR are not written consistent with the improved Standard TS (STS) format. The SE further stated that, when referencing the LTR in a licensing application, licensees should submit TSs that are consistent with their current approved TSs and the STS use and application section. The NRC staff's review of the proposed TS changes for PBAPS is discussed below.

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

The licensee proposed to revise the TS 3.3.1.1 required actions for Conditions I and J and to add a new Condition K. These changes support implementation of the Backup Stability Protection (BSP) requirements in the event that the OPRM Upscale RPS trip function (TS Table 3.3.1.1-1, Function 2.f) is inoperable. The proposed changes are consistent with the TS changes shown in Appendix B to the DSS-CD LTR, with the following exceptions:

- The proposed PBAPS TSs label the Required Actions as I.1, I.2, and I.3, instead of I.1, I.2.1, and I.2.2. The NRC staff finds these changes acceptable since they maintain consistency with the TS format discussed in Section 1.2, "Logical Connectors," of the PBAPS TSs.

- Proposed PBAPS Required Action I.2 references the "APRM Simulated Thermal Power – High Scram," instead of the "APRM flow-biased scram." The NRC staff finds this change acceptable since it uses terminology consistent with the title of this function in TS Table 3.3.1.1-1 (Function 2.b).

- Proposed PBAPS Required Action I.3 states "Initiate action to submit an OPRM report in accordance with Specification 5.6.8." The equivalent Required Action in the DSS-CD LTR (i.e., I.2.2) states, "Initiate action in accordance with Specification 5.6.6." The NRC staff finds the PBAPS changes to be acceptable since the required action is the same (i.e., initiate preparation of an OPRM report).

- Proposed PBAPS Required Action I.3 has a Completion Time of "Immediately" compared to a Completion Time of 90 days for the equivalent Required Action in the DSS-CD LTR (i.e., I.2.2). The PBAPS LAR provided proposed TS Bases for information only. With respect to this required action, the proposed TS Bases state:

  > This action should be initiated immediately to document the situation and prepare the report. The reporting requirements of Specification 5.6.8 document the corrective actions and schedule to restore the required channels to OPERABLE status. The Completion Time of 90 days shown in Specification 5.6.8 is adequate to allow time to evaluate the cause of the inoperability...

  The NRC staff finds the proposed changes to be more restrictive than the DSS-CD LTR TS requirements (i.e., immediately versus 90 days). In addition, the Completion Time of "Immediately" is consistent with PBAPS TSs 3.3.3.1 and 5.6.6 regarding

preparation of a post-accident monitoring report. Therefore, these changes are acceptable.

Based on the above, the NRC staff finds the proposed changes to TS 3.3.1.1, Required Actions I, J, and K to be acceptable, since they are either consistent with the DSS-CD LTR, or the exceptions to the LTR are acceptable.

TS Section 3.3.1.1, RPS Instrumentation, Surveillance Requirement (SR) 3.3.1.1.19

The licensee proposed to delete SR 3.3.1.1.19. The surveillance (which currently verifies that the OPRM is not bypassed) is no longer required because the DSS-CD automatically arms. The proposed change is consistent with the deletion of the equivalent SR in the DSS-CD LTR (SR 3.3.1.1.16). Therefore, the NRC staff finds the change acceptable.

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

The licensee proposed to revise the allowable value for TS Table 3.3.1.1-1, Function 2.b, "Simulated Thermal Power - High" (for two-loop operation) from ≤ 0.55W+ 63.3 percent RTP to ≤ 0.61W + 67.1 percent RTP (where "W" is the recirculation loop flow rate in percent of the design rating and "RTP" is rated thermal power). Note (b) with the Allowable Value for single loop operation remains unchanged. The basis for the change in allowable value is discussed in SAR Section 5.3.1, "APRM Flow-Biased Scram." The proposed change is intended to [[

]] The proposed change is consistent with the requirements specified in Section 8.0 of the DSS-CD LTR and is required for the implementation of DSS-CD. Based on the above, the NRC staff concludes the proposed change is acceptable.

TS Section 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1, Function 2.b, Note (g)

The licensee proposed to add new Note (g) to TS Table 3.3.1.1-1 associated with Function 2.b, which would read as follows:

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

As discussed in the table in Section 8.0 of the DSS-CD LTR, the proposed change is intended to reflect the change in the APRM allowable value due to implementation of the Automated BSP scram region.

This change is consistent with the requirements specified in Section 8.0 of the DSS-CD LTR. Therefore, the NRC staff finds the change acceptable.

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

Currently, TS Table 3.3.1.1-1, Function 2.f, "OPRM Upscale," requires that this function be operable under the following specified conditions: "≥ 23% RTP." The licensee proposes to change this requirement to "≥ 18% RTP."

Section 3.5 of the DSS-CD LTR states that the DSS-CD system is required to be operable in Mode 1 at all times. As an alternative, Section 3.5 further states that the DSS-CD may be required to be operable above a power level set at 5 percent of RTP below the lower boundary of the Armed Region defined by the MCPR monitoring threshold level. The LTR states that the alternate method is acceptable because system operability is assured prior to entry into the Armed Region.

As discussed in SAR Section 2.4.2, the OPRM Armed Region for PBAPS is defined as the region on the power/flow map with power ≥ 23.0 percent RTP and rated recirculation drive flow ≤ 75 percent. Therefore, the OPRM Upscale function must be operable at ≥ "18% RTP" (i.e., 23% - 5%).

The proposed change is consistent with the requirements specified in Section 3.5 of the DSS-CD LTR and the PBAPS plant-specific analyses. Therefore, the NRC staff finds the change acceptable.

TS Section 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1, Function 2.f, Notes (d) and (h)

The licensee proposed to delete Note (d) from TS Table 3.3.1.1-1, associated with Function 2.f. This note currently reads as follows.

> See COLR for OPRM period based detection algorithm (PBDA) setpoint limits.

In addition, the licensee proposed to add new Note (h) to TS Table 3.3.1.1-1 associated with Function 2.f, which would read as follows:

> 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.

With respect to the deletion of Note (d) as discussed in the table in Section 8.0 of the DSS-CD LTR, this note is no longer required since the PBDA will no longer be credited in the safety analyses.

With respect to the addition of Note (h) as discussed in the table in Section 8.0 of the DSS-CD LTR, the proposed requirement addresses the limited operability requirements during the initial testing phase following DSS-CD implementation.

The proposed changes are consistent with the requirements specified in Section 8.0 of the DSS-CD LTR. Therefore, the NRC staff finds the changes acceptable.

TS Section 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1, Function 2.f, SR 3.3.1.1.19

The licensee proposed to delete the reference to SR 3.3.1.1.19 from the list of required surveillances for the TS Table 3.3.1.1-1 OPRM Upscale function (Function 2.f).

As discussed above, SR 3.3.1.1.19 (which currently verifies that the OPRM is not bypassed) is being deleted. As such, the NRC staff concludes that deletion of the reference to SR 3.3.1.1.19 in TS Table 3.3.1.1-1 for Function 2.f is acceptable.

TS Section 3.4.1, Recirculation Loops Operating

The licensee proposed to revise the LCO for TS 3.4.1 to add the following note:

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

In addition, a new LCO Action would be added that would require immediate action to exit the MELLLA+ domain if the plant was operating in the MELLLA+ domain with a single recirculation loop in operation. This new Action would be designated as Action B, and the existing Action B would be re-designated as Action C.

Section 1.2.4 of the M+ LTR states that SLO is not allowed in the MELLLA+ operating domain. Accordingly, Limitation and Condition 12.5a of the NRC staff's SE for the M+ LTR states:

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

The NRC staff concludes that the proposed changes to TS 3.4.1 are consistent with Limitation and Condition 12.5a. Therefore, the proposed change is acceptable.

TS Section 5.6.3, Core Operating Limits Report (COLR)

TS 5.6.5, "Core Operating Limits Report (COLR)," requires that core operating limits be established and documented in the COLR for several specific items. One of these items, TS 5.6.5.a.5, currently reads as follows:

> The Oscillation Power Range Monitor (OPRM) Instrumentation for Specification 3.3.1.1.

The licensee has proposed to revise TS 5.6.5.a.5 to read as follows:

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

The proposed change to TS 5.6.5.a.5 will ensure that applicable thermal limits are reflected in the COLR following implementation of the DSS-CD stability solution. The proposed changes are consistent with the requirements specified in Section 8.0 of the DSS-CD LTR. Therefore, the NRC staff concludes that the proposed change is acceptable.

TS Section 5.6.8, OPRM Report

The licensee proposed to add new TS 5.6.8, "OPRM Report," which would read as follows:

> 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.

This new TS provides the requirements for submittal and content of the OPRM report consistent with Required Action I.3 in TS 3.3.1.1.

The proposed changes are consistent with the requirements specified in Section 8.0 of the DSS-CD LTR. Therefore, the NRC staff finds the changes acceptable.

TS Bases

Attachment 3 to the licensee's application dated September 4, 2014, provided revised TS Bases pages to be implemented with the associated TS changes. These pages were provided for information only and will be revised in accordance with the TS Bases Control Program discussed in TS 5.5.10.

5.0 TECHNICAL EVALUATION CONCLUSION

The NRC staff has reviewed the proposed PBAPS MELLLA+ amendment as discussed in SE Sections 3.0 and 4.0 above. The main conclusion from this review is that the broadening of the PBAPS operating domain by lowering the flow at high powers without additional limitations would reduce the safety margin, but the design and licensing basis changes proposed in the SAR are technically acceptable to maintain safety margin and satisfy the applicable regulatory criteria. The following changes are proposed to maintain similar safety margin under the MELLLA+ operating domain, as compared to under the current operating domain:

1) As discussed in SE Section 3.4.2, the OLMCPR increase will be comparable to 0.03 (relative to non-MELLLA+ operation) to account for the larger transient ΔCPR/ICPR calculated at the 83 percent flow condition.

2) As discussed in SE Section 3.2.2 (SAR Section 2.2.1), the SLMCPR will be increased from 1.10 and 1.09 for Unit 2 and Unit 3 respectively to 1.15 (including the penalty of +0.02 for operation above 42 MWt/Mlbm/hr penalty plus +0.03 to +0.04 due to use of SLO uncertainties penalty) for TLO.

3) As discussed in SE Section 4.1, a new license condition will prohibit operation in the MELLLA+ domain with a feedwater heater out of service resulting in more than a 10 °F reduction in feedwater temperature below the design feedwater temperature.

4) As discussed in SE Section 3.2.3 (SAR Sections 3.6.3) and SE Section 4.2, the TSs will be revised such that SLO will not be allowed in the MELLLA+ operating domain.

5) As discussed in SE Section 3.3.9 (SAR Sections 9.3.1 and 9.3.3), [[
]]

6) As discussed in SAR Section 2.4.1, [[
]]

7) As discussed in SE Section 3.3.9 (SAR Sections 9.3.1 and 9.3.3), the SLC system boron enrichment has been increased as part of the EPU amendment. This will reduce the integrated heat load to containment during ATWS under MELLLA+ conditions.

8) As discussed in SE Section 3.3.10 (SAR Section 10.6), new time-critical operator actions will be credited in the MELLLA+ licensing basis ATWS analysis such that the reactor water level will be reduced and SLC system boron injection will be initiated within 120 seconds of ATWS initiation.

As discussed in SE Section 3.6, the NRC staff concludes that the use of TRACG, for this application, is acceptable with the proposed EOP operator actions. Therefore, the applicable ATWS regulatory criteria (i.e., demonstrating that core coolability is maintained) are satisfied during ATWSI events for PBAPS. This staff review considered plant-specific information (e.g., EOPs); specific aspects of the TRACG code use as it was applied in the context of the PBAPS ATWSI analysis provided by the licensee (e.g., updates to the quench model; revision to the Tmin correlation in TRACG); and justification of the applicability of experimental data.

In the SRLR (Reference 2), the licensee stated: "The only differences between M+ SRLR for each unit will be typical unit-specific differences like the individual bundle designs and the core loading patterns; there are no differences related to the implementation of MELLLA+." During an NRC staff audit at PBAPS, the licensee confirmed that the transient analysis presented in the SAR and Unit 2 and Unit 3 Cycle 21 MELLLA + SRLR is based on bounding conditions for both units in MELLLA+ and increased core flow conditions.

Based on the considerations noted above and the discussion contained in SE Sections 3.0 and 4.0, the NRC staff concludes that the proposed MELLLA+ LAR for PBAPS is acceptable.

6.0 REGULATORY COMMITMENTS

The licensee did not make any regulatory commitments associated with this LAR.

7.0 RECOMMENDED AREAS FOR INSPECTION

As described above, the NRC staff conducted an extensive review of the licensee's plans and analyses related to the proposed MELLLA+ implementation and concluded that they are acceptable. The NRC staff's review of SAR Section 10.4, "Testing," identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed MELLLA+ operating domain expansion:

- 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 MCO magnitude and trend.

- 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 evaluate core thermal power, fuel thermal margin, and core flow performance to evaluate results against projected values and operational limits.

- A pressure regulator test will confirm that the pressure control system settings established for operation with the current power versus flow upper boundary at CLTP are adequate in the MELLLA+ operating domain.

- Reactor water level setpoint step changes will be introduced into the FW control system to verify that 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.

These areas are recommended based on the proposed testing for implementation of MELLLA+ at PBAPS, the extent and unique nature of changes necessary to implement the proposed MELLLA+ domain, and new conditions of operation necessary for operation in the proposed MELLLA+ domain. They do not constitute inspection requirements but are intended to give inspectors insight into important bases for the approval of the MELLLA+ LAR.

8.0 STATE CONSULTATION

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

9.0 PUBLIC COMMENTS

On December 2, 2014, the NRC staff published "Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving Proposed No Significant Hazards Considerations and Containing Sensitive Unclassified Non-Safeguards Information and Order Imposing Procedures for Access to Sensitive Unclassified Non-Safeguards Information" in the *Federal Register* associated with the proposed amendment request (79 FR 71450). In accordance with the requirements in 10 CFR 50.91, "Notice for public comment; State consultation," the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determination. Public comments were received regarding the proposed amendment (Reference 36). The comments received pertained to the PBAPS MELLLA+ LAR, as well as a MELLLA+ LAR for Grand Gulf Nuclear Station (GGNS) that was included in the same *Federal Register* notice. Some of the issues discussed in the public comments did not specifically pertain to the proposed NSHC determination. In addition, some of the issues discussed did not pertain to PBAPS, but to GGNS. The NRC staff addresses the remaining issue that both pertains to PBAPS and is within the scope of the proposed NSHC determination below.

The commentor stated, in part:

> You know very well that your statement that the MELLLA Plus changes involve no significant hazards consideration, for Grand Gulf and Peach Bottom is a brazen lie...
>
> In a letter less than a year ago regarding the Monticello Nuclear Reactor which is less than half the size of Grand Gulf it was pointed out: "Because MNGP has a small core with low power density, ATWS events with timely operator actions are predicted to cause cladding temperatures well below the regulatory limit... MELLLA+ applications with larger cores and higher power densities may result in instabilities that require the use of heat transfer models in TRACG04 for conditions that are still under NRC review."

NRC Response:

The NRC staff acknowledges that PBAPS, Units 2 and 3, have larger cores and higher power densities than Monticello. However, as discussed in SE Section 3.6, the NRC staff has reviewed the TRACG code models and concludes that TRACG calculations of ATWS with instability (ATWSI) events for PBAPS with possible rewetting and quenching is sufficient to provide reasonable assurance of compliance with the applicable ATWS regulatory criteria – namely, demonstrating that core coolability is maintained during ATWSI events. In addition, as discussed in SE Section 3.3.9 (SAR Sections 9.3.1 and 9.3.3), the licensee's TRACG calculations indicate that all applicable fuel limits are satisfied during ATWSI events and that the PCT for the most limiting ATWSI event is significantly lower that the criterion of 2200 °F.

10.0 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 SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve NSHC (79 FR 71454). The Commission received public comments pertaining to the proposed NSHC determination as discussed in SE Section 9.0. The NRC staff has reviewed the comments and determined that no modification to the proposed finding is necessary. 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.

11.0 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.

12.0 REFERENCES

1. Exelon letter to NRC dated September 4, 2014, "Peach Bottom Atomic Power Station, Units 2 and 3, License Amendment Request - Maximum Extended Load Line Limit Analysis Plus" (ADAMS Accession No. ML14247A503).

2. Exelon letter to NRC dated January 29, 2015, "Peach Bottom Atomic Power Station, Units 2 and 3, MELLLA+ License Amendment Request - Supplement 1 - Supplemental Reload Licensing Report" (ADAMS Accession No. ML15029A640).

3. Exelon letter to NRC dated February 6, 2015, "Peach Bottom Atomic Power Station, Units 2 and 3, MELLLA+ License Amendment Request - Supplement 2 - Response to Request for Additional Information" (ADAMS Accession No. ML15037A502).

4. Exelon letter to NRC dated April 28, 2015, "Peach Bottom Atomic Power Station, Units 2 and 3, MELLLA+ License Amendment Request - Supplement 3 - Response to Request for Additional Information" (ADAMS Accession No. ML15118A717).

5. Exelon letter to NRC dated July 6, 2015, "Peach Bottom Atomic Power Station, Units 2 and 3, MELLLA+ License Amendment Request - Supplement 4 - Response to Request for Additional Information" (ADAMS Accession No. ML15187A391).

6. Exelon letter to NRC dated September 4, 2015, "Peach Bottom Atomic Power Station, Units 2 and 3, MELLLA+ License Amendment Request - Supplement 5 - Response to Request for Additional Information" (ADAMS Accession No. ML15247A088).

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

8. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-33075P-A, Revision 8, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density," dated November 2013 (ADAMS Accession No. ML13324A097).

9. GE Hitachi Nuclear Energy, Licensing Topical Report NEDE-33147P-A, Revision 4, "DSS-CD TRACG Application," dated August 2013 (ADAMS Accession No. ML13224A319).

10. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-33173P-A, Revision 4, "Applicability of GE Methods to Expanded Operating Domains," dated November 2012 (ADAMS Accession No. ML123130130).

11. GE Nuclear Energy, Licensing Topical Report, NEDE-32906P-A, Revision 3, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," dated September 2006 (ADAMS Accession No. ML062720163).

12. GE Nuclear Energy, Licensing Topical Report, NEDE-32906P, Supplement 1-A, "TRACG Application for Anticipated Transient Without Scram Overpressure Transient Analyses," dated November 2003 (ADAMS Accession No. ML033381073).

13. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDE-32906P, Supplement 3-A, Revision 1, "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated April 2010 (ADAMS Accession No. ML110970401).

14. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-33173, Supplement 3P-A, Revision 1, "Applicability of GE Methods to Expanded Operating Domains -Supplement for GNF2 Fuel," dated July 2011 (ADAMS Accession No. ML111960462).

15. NRC, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001, Revision 0, dated December 2003 (ADAMS Accession No. ML033640024).

16. NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, dated September 30, 2007 (ADAMS Accession No. ML072640413).

17. NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, dated November 2012 (ADAMS Accession No. ML12324A013).

18. NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980 (ADAMS Accession No. ML051400209).

19. Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 4, dated March 2012 (ADAMS Accession No. ML111330278).

20. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, dated May 2011 (ADAMS Accession No. ML100910006).

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

22. NRC Staff Requirements Memorandum for SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated July 21, 1993 (ADAMS Accession No. ML003708056).

23 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." Relevant sections used for the review of this LAR are as follows:

    (a) Chapter 4, Section 4.2, "Fuel System Design," Revision 3, dated March 2007 (ADAMS Accession No. ML070740002).

(b) Chapter 4, Section 4.3, "Nuclear Design," Revision 3, dated March 2007 (ADAMS Accession No. ML070740003).

(c) Chapter 4, Section 4.4, "Thermal and Hydraulic Design," Revision 2, dated March 2007 (ADAMS Accession No. ML070550060).

(d) Chapter 4, Section 4.6, "Functional Design of Control Rod Drive System," Revision 2, dated March 2007 (ADAMS Accession No. ML070540139).

(e) Chapter 5, Section 5.2.2, "Overpressure Protection," Revision 3, dated March 2007 (ADAMS Accession No. ML070540076).

(f) Chapter 5, Section 5.4.6, "Reactor Core Isolation Cooling System (BWR)," Revision 4, dated March 2007 (ADAMS Accession No. ML070540102).

(g) Chapter 5, Section 5.4.7, "Residual Heat Removal (RHR) System," Revision 5, dated May 2010 (ADAMS Accession No. ML100680577).

(h) Chapter 6, Section 6.2.1.1.C, "Pressure Suppression Type BWR Containments," Revision 7, dated March 2007 (ADAMS Accession No. ML063600403).

(i) Chapter 6, Section 6.2.1.2, "Subcompartment Analysis," Revision 3, dated March 2007 (ADAMS Accession No. ML070620009).

(j) Chapter 6, Section 6.2.2, "Containment Heat Removal Systems," Revision 5, dated March 2007 (ADAMS Accession No. ML070160661).

(k) Chapter 6, Section 6.2.5, "Combustible Gas Control In Containment," Revision 3, dated March 2007 (ADAMS Accession No. ML070620006).

(l) Chapter 7, Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems," Revision 6, dated July 2012 (ADAMS Accession No. ML110550791).

(m) Chapter 9, Section 9.3.5, "Standby Liquid Control System (BWR)," Revision 3, dated March 2007 (ADAMS Accession No. ML070680186).

(n) Chapter 15, Section 15.0, "Introduction – transient and Accident Analyses," Revision 3, dated March 2007 (ADAMS Accession No. ML070710376).

(o) Chapter 15, Section 15.8, "Anticipated Transients Without Scram," Revision 2, dated March 2007 (ADAMS Accession No. ML070570008).

(p) Chapter 15, Section 15.9, "Boiling Water Reactor Stability," dated March 2007 (ADAMS Accession No. ML070550017).

(q) Chapter 18, Section 18.0, "Human Factors Engineering," Revision 2, dated March 2007 (ADAMS Accession No. ML070670253).

(r) Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007 (ADAMS Accession No. ML071700658).

24. Global Nuclear Fuel, "General Electric Standard Application for Reactor Fuel," NEDE 24011P-A and NEDE-24011P-A-US (latest approved revision).

25. Exelon letter to NRC dated December 5, 2014, "Peach Bottom Atomic Power Station, Unit 2, License Amendment Request – Safety Limit Minimum Critical Power Ratio Change" (ADAMS Accession No. ML14342A229).

26. Exelon letter to NRC dated April 30, 2015, "Peach Bottom Atomic Power Station, Unit 3, License Amendment Request – Safety Limit Minimum Critical Power Ratio Change" (ADAMS Accession No. ML15120A290).

27. Exelon letter to NRC dated October 1, 2015, "Peach Bottom Atomic Power Station, Units 2 and 3, MELLLA+ License Amendment Request - Supplement 6 - Response to Request for Additional Information" (ADAMS Accession No. ML15274A467).

28. NRC Generic Letter 1989-10, "Safety Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989 (ADAMS Accession No. ML031150300).

29. NRC Generic Letter 1989-16, "Installation of a Hardened Wetwell Vent," dated September 1, 1989 (ADAMS Accession No. ML031140220).

30. NRC Generic Letter 1995-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," dated August 17, 1995 (ADAMS Accession No. ML031070145).

31. NRC Generic Letter 1996-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions," dated September 30, 1996 (ADAMS Accession No. ML031110021).

32. NRC Generic Letter 1989-08, "Erosion/Corrosion-Induced Pipe Wall Thinning, dated May 2, 1989 (ADAMS Accession No. ML031200731).

33. NUREG-0661, "Mark I Containment Long-Term Program Safety Evaluation Report," dated July 1980 (ADAMS Accession No. ML11203A031).

34. Exelon letter to NRC dated February 15, 2013, "Peach Bottom Atomic Power Station, Units 2 and 3, Supplemental Information and Corrections Supporting Request for License Amendment Request - Extended Power Uprate - Supplement No. 1" (ADAMS Accession No. ML13051A032).

35. GE-Hitachi Nuclear Energy letter to NRC dated August 26, 2015, "Clarification of Limitation and Condition 23 for NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains" (ADAMS Accession Number ML15238A687).

36. Docket No. NRC-2014-0250, Comment (1) of Unknown Individual on Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving Proposed No Significant Hazards Considerations and Containing Sensitive Unclassified Non-Safeguards Information and Order Imposing Procedures for Access to Sensitive Unclassified Non-Safeguards Information, dated January 2, 2015 (ADAMS Accession No. ML15009A028).

37. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-32983P-A, Revision 2, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," dated January 2006 (ADAMS Accession No. ML072480121).

38. Exelon letter to NRC dated October 26, 2015, "Peach Bottom Atomic Power Station, Units 2 and 3, PBAPS MELLLA+ License Amendment Request - Supplement 7- Revision Regarding Proposed Technical Specifications" (ADAMS Accession No. ML15299A084).

39. Exelon letter to NRC dated January 15, 2016, "Peach Bottom Atomic Power Station, Unit 2 and Unit 3, MELLLA+ License Amendment Request - Supplement 8 - Proprietary Labeling Corrections and Implementation Plan Change" (ADAMS Accession No. ML16019A101).

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Appendices:
A. Request for Additional Information Evaluation
B. List of Acronyms

APPENDIX A
REQUEST FOR ADDITIONAL INFORMATION EVALUATION

This appendix provides a summary of the NRC staff's evaluation of the licensee's responses to Reactor Systems Branch (SRXB) requests for additional information (RAIs) SRXB-RAI-1 through SRXB-RAI-18. The RAI responses were contained in the following licensee submittals:

- SRXB-RAI-1 through SRXB-RAI-17: Letter dated April 28, 2015 (Reference 4)
- SRXB-RAI-18: Letter dated October 1, 2015 (Reference 27)

<u>SRXB-RAI-1, Power density > 50 MWt/Mlbm/hr</u>

> Appendix A to the SAR lists the limitations and conditions listed in Section 9.0 of the NRC staff safety evaluation (SE) for GE-Hitachi Nuclear Energy Americas LLC (GEH) licensing topical report (LTR) NEDC-33173P-A (ADAMS Accession No. ML121150469), referred to as the Methods LTR. Limitation and Condition 9.3 reads as follows:
>
> > Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to core flow ratio will not exceed 50 MWt/Mlbm/hr at any statepoint in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWt/Mlbm/hr, the application will provide power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased.
>
> The power distribution root mean square (RMS) data provided to support the Methods SE (Method LTR Figure 3-4) ranged from [[ ]] and an extrapolation to 50 MWt/Mlbm/hr was allowed based on the safety limit minimum critical power ratio (SLMCPR) adders.
>
> As discussed in Section 1.2.1 and 2.2.5 of the SAR, and shown in Table 2-3 of the SAR, the power-to-flow ratio at the low flow/high power statepoint "K" (55 percent of core flow, 78.8 percent of current licensed thermal power) is 55.23 MWt/Mlbm/hr, which exceeds the 50 MWt/Mlbm/hr value in Limitation and Condition 9.3. As such, a power distribution assessment is required. Provide a copy of a recent traversing incore probe (TIP) report and an evaluation of the power distribution uncertainties in PBAPS, showing historical power distribution uncertainties as function of burnup, to demonstrate that PBAPS is not an outlier plant (compared to other plants in the fleet).

<u>Resolution of SRXB-RAI-1</u>

The licensee provided comparisons of TIP data versus PANACEA calculations that show that, in spite of the large power density, the power distribution uncertainties in Peach Bottom Atomic Power Station (PBAPS) are within acceptable limits. The TIP comparison with calculations indicates that for the typical condition, the power distribution uncertainty is between 2.5 and 8 percent, depending on the type of uncertainty (see Table A-1). Table A-1 shows that PBAPS power distribution uncertainty levels are not unusual when compared to the rest of the fleet.

Note that these uncertainties are the ones that have been measured during extended power uprate (EPU) operation in PBAPS (up to 44.07 MW/Mlb/hr) and their impact on the SLMCPR is already reflected in the current values. The expectation is that an increase from 42 to 55 MWt/Mlbm/hr would increase these uncertainties by a relatively small amount, which should be covered by the 0.02 SLMCPR penalty.

In the TIP report, the licensee has also provided a study to attempt to identify any trend in the power distribution uncertainty with respect to power-flow ratio. None was observed, confirming that the methodology should be applicable to the higher power-flow ratios in MELLLA+

Table A-1 - Average TIP-PANACEA uncertainties showing that PBAPS is not at outlier

	TIP Type	Radial TIP RMS	Axial TIP RMS	Nodal TIP RMS
Plant A	Neutron	[[		
Plant B	Neutron			
PBAPS 2	Gamma			
PBAPS 3	Gamma			]]

The licensee provided the required plant-specific power distribution uncertainty evaluation. The NRC staff determined that the supplemental information demonstrates that PBAPS is not an outlier plant and that the neutronic methods axial and nodal power distribution uncertainties have not increased; therefore, this RAI is closed,

SRXB-RAI-2, SLMCPR Adders

SAR Section 2.2.1, "Safety Limit Minimum Critical Power Ratio," states that "[t]he cycle-specific SLMCPR analysis will incorporate a +0.02 SLMCPR adder for MELLLA+ operation." Section 2.2.2 "Operating Limit Minimum Critical Power Ratio [OLMCPR]," states that "[w]ith the usage of TRACG-AOO instead of ODYN the +0.01 adder to the resulting OLMCPR as required by Methods LTR SER Limitation and Condition 9.19 is no longer applicable and will not be applied to the OLMCPR."

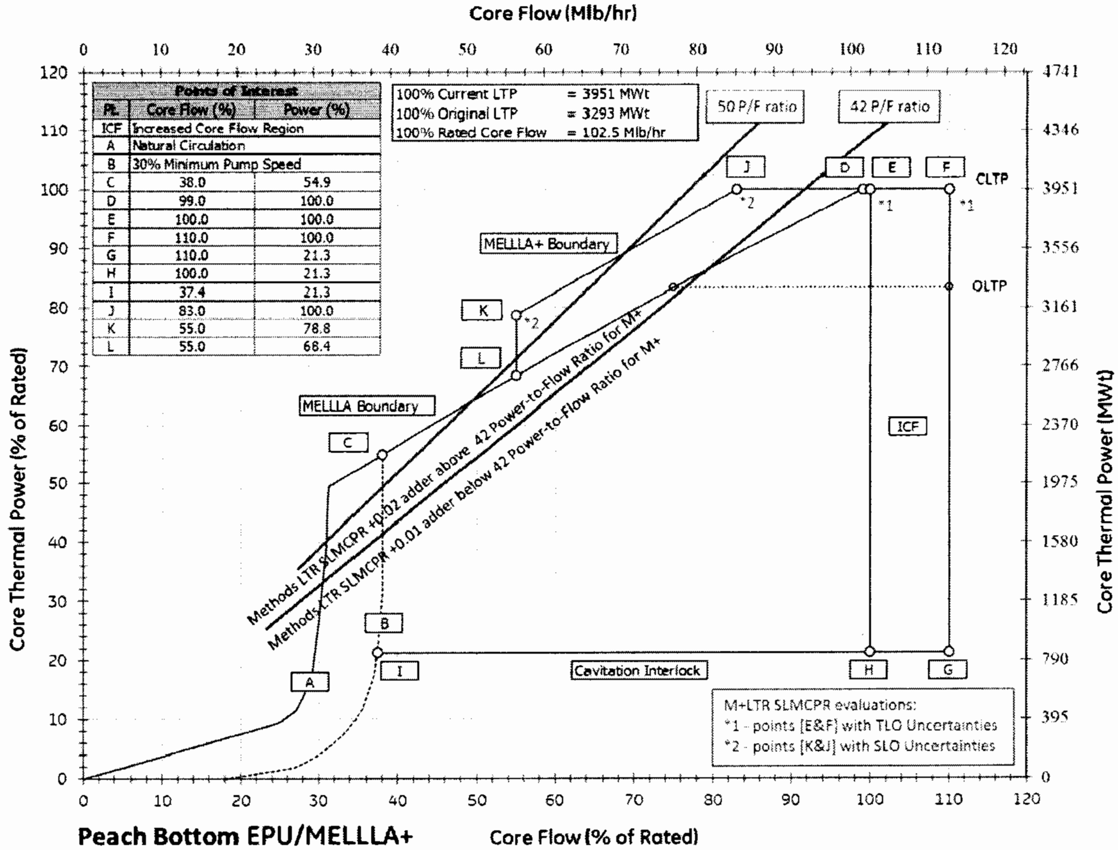
Provide a list of SLMCPR and OLMCPR adders in MELLLA+ with respect to pre-EPU conditions. Specify which adders are part of the EPU upgrade, and which are MELLLA+ specific.

Resolution of SRXB-RAI-2

The licensee provided an illustration of the SLMCPR adder used as show in Figure A-1. For points J and K, which lie inside the MELLLA+ region, single loop operation (SLO) uncertainties are used. For Points F, E, D, and L, two loop operation (TLO) uncertainties are used. For all points, the Methods +0.02 adder is used conservatively, even though a value of +0.01 could have been used for points F, E, and D because they lie under the 42 MWt/Mlbm/hr line.

Conservative application of the +0.02 adder is acceptable because it will establish the most limiting SMCPR values in the Technical Specifications. This RAI is closed.

Figure A-1 - Illustration of SLMCPR adders



SRXB-RAI-3, FW Temperature

SAR Section 2.4.4 "M+LTR SER Limitation and Condition 12.5.b," states that feedwater (FW) temperature will be limited to be greater than 371.5 °F.

Further, SAR Section 2.4.1 states:

[[

]]

a) Provide the variability in FW temperature for a typical 24-month cycle at full power conditions (i.e., min/max for the cycle).

b) Provide a justification for how [[

]].

c) Expand on the reasoning why indefinite operation at reduced FW temperature is acceptable in the M+ domain, including what analysis was done to support this conclusion.

d) Provide plots of CPR versus time, and the time of DSS-CD scram for the TRACG calculations described in M+ SAR Section 2.4.1 [[ ]].

Resolution of SRXB-RAI-3

The licensee provided historical full power operating data. The FW temperature for Cycle 20 oscillated at most approximately 2 °F from nominal, which justifies the application of the ±10 °F criterion.

The licensee states that the analysis in Section 2.4.1 was performed [[ ]]

The plot of critical power ration (CPR) vs time [[ ]] was provided [[ ]] This RAI is closed.

SRXB-RAI-4, Maximum Core Flow

The power-flow map (SAR Figure 1-1) shows Point F with 110% flow in the increased core flow (ICF) region. What is the maximum core flow that PBAPS can achieve? Is this a function of exposure (e.g., bottom-peaked shapes may result in reduced max achievable flow)? Is PBAPS susceptible to bi-stable flow in the recirculation loops? If so, what is the maximum (or range of) achievable recirculation flow used in normal operation to minimize bi-stable flow concerns?

Resolution of SRXB-RAI-4

PBAPS can reach 110 percent flow near end-of-cycle (EOC) when the core pressure drop is lower as a result of the top peaked power shape. Thus, PBAPS can use increased flow to stretch the cycle life.

PBAPS has observed small flow oscillations due to bistable flow, but the effect is not significant and causes no relevant restrictions. This RAI is closed.

SRXB-RAI-5, OLMCPR and SLMCPR Values

In reference to SAR Tables 2-4 and 2-5, provide the calculated MCPR margin for the equilibrium M+ cycle. Provide the OLMCPR and SLMCPR values (both two-loop operation (TLO) and single loop operation (SLO)) for the current operating cycle at PBAPS even if not designed for MELLLA+ operation.

Resolution of SRXB-RAI-5

The licensee provided SLMCPR and OLMCPR values in Section 11 of the SRLR. The NRC staff confirmed that the detect and suppress solution – confirmation density (DSS-CD) amplitude discriminator setpoint (S_{AD}) acceptability criteria from Tables 2-4 and 2-5 of the SAR are met for TLO and SLO. This RAI is closed.

SRXB-RAI-6, Exit Void Fraction

> On SAR Figures 2-2 through 2-6, explain the difference between the lines labeled "PBAPS M+SAR" and "PBAPS M+SAR 100F." Do these refer to points D and J in Figure 1.1? What is the equivalent operating point for the other plants (A, B, C, D, E, and F) shown in Figures 2-2 through 2-6?
>
> The text in SAR Section 2.1.2 "Core Design and Fuel Thermal Monitoring Threshold." states:
>
> > Figures 2-3 through 2-5 shows [sic] that exit voiding at PBAPS is higher than other plants. This is because of operating a high power density plant at lower CFs [core flows] through the entire cycle.
>
> Are the other plants in these figures operated with the planned flow as a function of exposure, or at 100 percent flow?

Resolution of SRXB-RAI-6

"PBAPS M+SAR" refers to the planned average flow operation, which is approximately 90 percent flow. "PBAPS M+SAR 100F" refers to a calculated condition at 100 percent flow for reference only.

The comparison provided against other plants is based on actual operating flow, which is typically lower than 100 percent and, thus, results in higher void fraction. This RAI is closed.

SRXB-RAI-7, Backup Stability Solution

> SAR Section 2.4.3 "Backup Stability Protection," describes that the detect and suppress solution - confirmation density (DSS-CD) LTR provides two options: (1) backup stability protection (BSP) manual regions and (2) BSP implemented with average power range monitor (APRM) flow-biased scram. This section of the PBAPS SAR appears to be a summary of the DSS-CD LTR, but it is not clear which of the two options will be implemented by PBAPS. Which option will PBAPS use for the first MELLLA+ cycle?
>
> Have the BSP regions been evaluated for the PBAPS equilibrium cycle? Provide them if available. If not, where will they be documented?

Resolution of SRXB-RAI-7

Both BSP options are used consistent with the proposed changes to PBAPS Technical Specification 3.3.1.1, Conditions I and J. In case DSS-CD becomes inoperable, the licensee will immediately implement the Manual BSP regions solution. Within at most 12 hours, PBAPS switches to automated BSP after they have had time to implement the required changes in the reactor protection system. If the automated BSP is subsequently declared inoperable or is not implemented within 12 hours of DSS-CD becoming inoperable, then the licensee will reduce operation to below the BSP boundary defined in the core operating limits report (COLR) (by reducing power or increasing flow) plus the BSP regions remain in effect.

With respect to the BSP regions, the licensee stated that the regions and the BSP boundary have been established for PBAPS, Unit 2, Cycle 21 on Figures 16 and 17 in the Supplemental Reload Licensing Report (SRLR) (Reference 2). Similar information will be developed and included in the SRLR for Unit 3.

This RAI is closed.

SRXB-RAI-8, Plant Design Parameters

> Provide additional plant design parameters relevant to the ATWS calculations in Section 9 of the SAR. Specifically: turbine bypass capacity, sources of high pressure injection and their operability issues (e.g., steam is lost after isolation), sources of low pressure injection and their operability issues (e.g., condensate storage tank (CST) pumps). Are FW pumps steam driven, or motor driven? Provide vessel component elevations in units comparable to the ones used for water level in the Section 9 figures (e.g., separators, FW spargers, nominal level, level setpoints for actuations, and top of active fuel (TAF)).

Resolution of SRXB-RAI-8

The licensee provided the requested information. Turbine bypass capacity is 17.4 percent. No sources of low pressure injection are credited. Only high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) are credited for isolation anticipated transient without scram (ATWS), and they use suction from the condensate storage tank. For the non-isolated case (turbine trip with bypass), the FW pumps, which are steam driven remain available.

For PBAPS, the HPCI and RCIC pumps are steam driven, but the steam is supplied from inside containment via their own penetration, and it is not affected by containment isolation. Vessel elevations were provided.

This RAI is closed.

SRXB-RAI-9, ATWS Sequence of Events

> Provide tables of the assumed sequence of events for the ODYN licensing calculation and the ATWS/instability calculation. Describe the sources of water used to control the reactor level. For the equipment used, describe automated

actions and other assumptions about operability after the main steam isolation valve (MSIV) isolation occurs.

<u>Resolution of SRXB-RAI-9</u>

PBAPS uses ODYN as the licensing ATWS calculation and uses TRACG for ATWS over-pressure calculations. The response to SRXB-RAI-9 provides the sequence of events for both ODYN isolation ATWS, TRACG ATWS, and pressure regulator failure open (PRFO) overpressure events. The NRC staff finds that the appropriate assumptions were used in the licensee's analysis; therefore, this RAI is closed.

<u>SRXB-RAI-10, Operator Critical Actions</u>

> Typically, critical operator actions for ATWS include: place the reactor switch in shutdown mode, initiate reduction of water level, initiate standby liquid control system (SLCS) injection, and terminate and prevent injection into the core. Provide a table with the critical operator actions in PBAPS, including two columns: (a) required timing by TS/procedure/training, and (b) assumed timing in the ATWS calculations. Note: report "NA" on the calculation column if the operator critical action has no impact on the calculation (e.g., place switch in shutdown mode).

<u>Resolution of SRXB-RAI-10</u>

The RAI response provides a table with the operator critical action timing assumed in the analysis for four events. The critical operator actions are:

1. Place mode switch in Shutdown
2. Initiate SLC system injection
3. Terminate and prevent reactor pressure vessel (RPV) injection to lower reactor water level
4. Initiate residual heat removal (RHR) containment cooling

The time critical operator action timing is defined in the licensee's operator response time program and ensured during training. The NRC staff finds that appropriate assumptions were used in the analyses for operator actions; therefore, this RAI is closed.

<u>SRXB-RAI-11, ATWS with Instability (ATWSI) Figures</u>

> The neutron flux provided for the ATWS-instability recirculation pump trip (RPT) calculation is core-average, and the power oscillations [[
>
>]] Is the oscillation out-of-phase (OOP)? Provide additional plots with hot channel powers at symmetric core locations showing the amplitude of the regional oscillations for the ATWS-instability calculation.

Resolution of SRXB-RAI-11

Additional plots are provided in the RAI response. The oscillations are out of phase with large amplitude oscillations. This RAI is closed.

SRXB-RAI-12, ATWSI Calculation Details

a) Section 9.3.3 of the SAR specifies that [[ ]] Is the TRACG quench model turned on for these calculations? Is it activated for the ATWS/instability transient?

b) The ATWS/instability calculation (SAR Figure 9-8) shows [[ ]] What mechanism allows for rewet if the quench model is turned off?

c) Provide plots similar to SAR Figure 9-8 that shows PCT superimposed with the calculated Tmin value.

Resolution of SRXB-RAI-12

As part of the response to SRXB-RAI-12 in Reference 4, the licensee submitted a comparison of calculated peak cladding temperature (PCT) versus minimum temperature for stable film boiling (Tmin) during an ATWSI calculation. Middle of cycle (MOC) conditions are presented with a turbine trip with bypass (TTWBP) transient and failure to scram. As seen in Figures A-2 and A-3 below, the TTWBP cases using the Modified Shumway Tmin correlation [[ ]]

Figure A-2 - [[

]]

[[

]]

Figure A-3 - [[

]]

[[

]]

The quench model is turned on for the ATWSI analysis, but it is not exercised in the cases presented in response to SRXB-RAI-12, which used the Modified Shumway Tmin correlation because [[ ]] This can also been seen from the comparison of the case with the quench model turned on and off in Figure A-3. Plots of PCT with and without quench model

illustrate this point. [[

]] As the power generation is reduced by the actions of the water level reduction and boron injection, the oscillations diminish and the heat transfer remains in nucleate boiling and the cladding temperature remains near the saturation temperature. A plot of PCT along with calculated Tmin is provided showing [[ ]] The NRC staff finds that this is acceptable for this calculation because the quench model is not used and the cladding temperature [[ ]] Therefore, this RAI is closed.

Note that the evaluation of SRXB-RAI-18 addresses the relative conservatism of the use of the Modified Shumway versus the Homogeneous Nucleation Temperature correlation for Tmin during ATWSI events. This was not the purpose of SRXB-RAI-12.

SRXB-RAI-13, Fluence Calculations

> What methods are used for PBAPS fluence calculations for the Pressure and Temperature Limits Report (PTLR) results? How is the uncertainty in the fluence calculation impacted by MELLLA+? Does the uncertainty stay below 20%?

Resolution of SRXB-RAI-13

Fluence is calculated based on the NRC-approved methodology in licensing topical report (LTR) NEDC-32983P-A, Revision 2 (Reference 37). The licensee confirmed that a mix of methods is not used. The licensee concluded that uncertainty under MELLLA+ conditions [[ ]] and the uncertainty is less than 20 percent. The licensee specifically evaluated the effect of MELLLA+ operation on reactor vessel fast neutron fluence and [[ ]] This RAI is closed.

SRXB-RAI-14, N/A

Question SRXB-RAI-14 was withdrawn by the NRC staff because it was not applicable.

Resolution of SRXB-RAI-14

This RAI is no longer applicable.

SRXB-RAI-15, DSS-CD Calculation

> SAR Section 2.4.1, "DSS-CD Setpoints," page 2-13, states:
>
> [[
>
>]]

Provide the TRACG transient results to demonstrate that [[ ]] In the results, mark the time that DSS-CD detects the oscillations and would have scrammed if active for this calculation.

Resolution of SRXB-RAI-15

The calculation results were provided by the licensee. The DSS-CD trip occurs at approximately [[ ]] seconds into the transient, which provides [[ ]] seconds of margin until the SLMCPR is challenged. The NRC staff recognizes that for the data provided, the reactor is assumed to operate at OLMCPR limits; realistically, the initial minimum critical power ratio (IMCPR) would be larger and additional margin would exist. The staff finds that the DSS-CD calculation provides appropriate margin to the SLMCPR; therefore, this RAI is closed.

SRXB-RAI-16, Peak Pressure

With regard to SAR Section 3.1.2, "Overpressure Relief Capacity," what is the peak calculated pressure for the overpressure analyses? Specify the most limiting overpressure event.

Resolution of SRXB-RAI-16

The requested information was provided by the licensee. The licensee stated, in part, that:

> ... the limiting overpressure event is the main steam isolation valve closure with scram on high flux (MSIVF). The resulting peak dome pressure is 1324 psig and the peak RPV pressure is 1352 psig. Both are less than the ASME Service Level B limit of 1375 psig.

The NRC staff finds that the results of the licensee's analysis demonstrate that the American Society for Mechanical Engineers (ASME) Service Level B limit is satisfied; therefore, this RAI is closed.

SRXB-RAI-17, Void vs Bundle Power Outliers

There are three outliers that are visually apparent in SAR Figure 2-17 at Integrated Bundle Power vs Bundle Average In-Channel Void Fraction of about 0.5 and 0.41, respectively. Provide a discussion on these and any other outliers in Figure 2-17.

Resolution of SRXB-RAI-17

The licensee clarified that the points that were described as "outlier points" correspond to periphery channels with very low power. The clarification resolved the NRC staff's concern; therefore, this RAI is closed.

SRXB-RAI-18, Minimum Stable Film Boiling Temperature Sensitivities

Note: This RAI question was prepared by the NRC staff as follow up to an audit at GE-Hitachi Nuclear Energy (GEH) from August 31 through September 2, 2015. The main purpose of the

audit was to review sensitivity calculations and methodologies for ATWSI performed using GEH evaluation model TRACG. The response to this RAI was provided in Exelon's letter dated October 1, 2015 (Reference 27). The RAI question was as follows:

> The NRC staff needs to better understand the performance of TRACG beyond Nucleate Boiling (i.e., in Transition Boiling and Film Boiling). The staff has questions regarding the use of a Minimum Film Boiling Temperature (Tmin) model in TRACG as a means of predicting the transition to film boiling and determining the Transition and Film Boiling Heat Transfer Coefficient.
>
> 1) Provide TRACG turbine trip with bypass (TTWBP) and dual recirculation pump trip (2RPT) sensitivity calculations for PBAPS in which the Homogeneous Nucleation Temperature is used for Tmin; include relevant plots of results.
>
> 2) Additionally, provide TRACG sensitivity calculations with more realistic assumptions in which both the Homogenous Nucleation Temperature and the Shumway correlation (as currently implemented in TRACG) is used. Include relevant sensitivity parameters (such as, but not necessarily limited to: operator response time, Tmin model, peak cladding temperature (PCT) and/or number of assemblies that exceed 2200 °F, peaking factor and feedwater assumptions); also include relevant plots of results.

<u>Resolution of SRXB-RAI-18</u>

Based on information from independent test data relevant to ATWSI calculations, the NRC staff needed to learn more about the impact of modeling assumptions and correlations on ATWSI calculations. Specifically, based on this experimental data, the staff believes that there is uncertainty on the appropriateness of using the Modified Shumway correlation for Tmin during thermal-hydraulic instabilities and may yield non-conservative results. For this reason, the staff requested that sensitivity studies be performed using the Homogenous Nucleation Temperature (THN), which the staff believes to be conservative.

The licensee provided sensitivities to:

1. Feedwater cooling rate. Two cooling rates were used for the analysis:

    a. A licensing case, where a 60 second exponential time constant is used to reduce feedwater temperatue from nominal to condenser temperature.

    b. A nominal case based on training simulator modeling where the cooling rate of 1.5°F/second is assumed following 20 seconds of the temperature remaining constant, which results in a slower cooling rate (see Figure A-4).

2. Operator action timing:

    a. 90 seconds, 120 seconds, and 200 seconds to initiate feedwater flow reduction.

3. Hot rod peaking factor:

a. [[ ]], corresponding to 95 percent of the operating limit.

b. [[ ]], corresponding to the actual PBAPS peaking factor at the exposure analyzed.

4. Tmin model. Two models were used:

   a. Modified Shumway Correlation.

   b. THN, which is significantly more conservative.

The licensee provided the requested sensitivity calculations. Tables 1 and 2 of the RAI response (reproduced here, in part, as Tables A-2 and A-3) summarize the PCT results of TTWBP-ATWSI sensitivity calculations.

Figure A-4 - Turbine Trip Feedwater Temperature Reduction

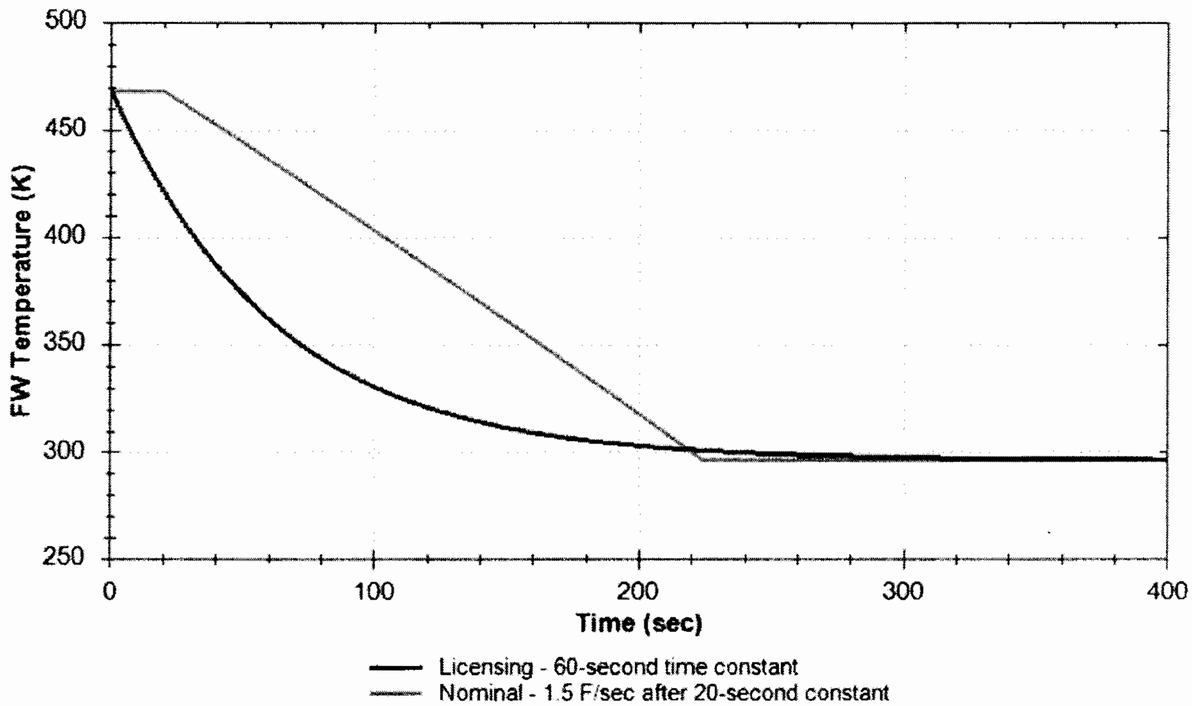


Table A-2 - TTWBP-ATWSI Sensitivities PCT Results

Case Figure #	Transient FW Temperature Response	Operator Action Time to Reduce Water Level (sec)	Tmin Model	Limiting Rod Peaking (kW/ft)	PCT (K)
RG, Figure A-5	Licensing	120	Modified Shumway	95% limit = [[ ]]	[[ ]]
RG4, Figure A-6	Nominal	120	Modified Shumway	95% limit = [[ ]]	[[ ]]
RG6, Figure A-7	Nominal	120	Homogeneous Nucleation	95% limit = [[ ]]	[[ ]]

Notably, in the licensing basis calculations the licensee used a conservative feedwater (FW) temperature for turbine trip events. When the licensee used a nominal (i.e., realistic) FW temperature transient this delayed the onset of instabilities as expected and resulted in no transition boiling after the initial pressure and power spike for the 120 second water level reduction cases. Thus, the sensitivities highlight the impact of the timing of the operator action to reduce water level relative to the onset of instabilities that result in a cladding temperature excursion.

The sensitivities also demonstrate the impact of the Tmin model used in cases where there is a cladding temperature excursion. Using the more conservative Homogeneous Nucleation Temperature [[ ]]

[[ ]] Since ATWS is a beyond design-basis event, best estimate nominal conditions should be used for the analysis; therefore, the NRC staff concludes that the ATWSI acceptance criteria are satisfied for PBAPS.

Table A-3 - 2RPT-ATWSI Sensitivities PCT Results

Case, Figure #	Operator Action Time to Reduce Water Level (sec) (Includes time for manual scram)	Tmin Model	Limiting Rod Peaking (kW/ft)	PCT (K)
RG1a, Figure A-8	180	Modified Shumway	95% limit = [[ ]]	[[ ]]
RG4a, Figure A-9	180	Homogeneous Nucleation	95% limit = [[ ]]	[[ ]]
RG5a, Figure A-10	150	Homogeneous Nucleation	95% limit = [[ ]]	[[ ]]
RG8, Figure A-11	150	Homogeneous Nucleation	95% limit = [[ ]]	[[ ]]

The licensee conducted sensitivities for the 2RPT transient. In cases RG4a, RG5a, and RG9 the licensee performed 2RPT calculations using THN for Tmin. [[

]] For the worst-case scenario, [[

]] thus, the NRC staff concludes that ATWSI acceptance criteria of maintaining a coolable core geometry is satisfied. This RAI is closed.

Figure A-5
TTWBP, Licensing FW Temperature, 120-second Operator Action to Reduce Water Level, Modified Shumway Tmin

[[

]]

Figure A-6
TTWBP, Best-Estimate FW Temperature, 120-second Operator Action to Reduce Water Level, Modified Shumway Tmin

[[

]]

Figure A-7
TTWBP, Best-Estimate FW Temperature, 120-second Operator Action to Reduce Water Level, Homogeneous Nucleation Tmin

[[

]]

Figure A-8
2RPT, 180-second Operator Action to Reduce Water Level, Modified ShumwayTmin, Initial LHGR at 95% of Limit

[[

]]

Figure A-9
2RPT, 180-second Operator Action to Reduce Water Level, Homogeneous Nucleation Tmin, Initial LHGR at 95% of Limit

[[

]]

Figure A-10
2RPT, 150-second Operator Action to Reduce Water Level, Homogeneous Nucleation Tmin, Initial LHGR at 95% of Limit

[[

]]

Figure A-11
2RPT, 150-second Operator Action to Reduce Water Level, Homogeneous Nucleation Tmin, Initial LHGR at Core Maximum

[[

]]

APPENDIX B
LIST OF ACRONYMS

ACRONYM	DEFINITION
2RPT	Two-Pump Recirculation Pump Trip
ΔCPR	Delta Critical Power Ratio
ABA	Amplitude Based Algorithm
ABSP	Automated Backup Stability Protection
AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AL	Analytical Limit
ALARA	As Low as is Reasonably Achievable
AOO	Anticipated Operational Occurrences
AOPs	Abnormal Operating Procedures
APRM	Average Power Range Monitor
ARI	Alternative Rod Insertion
ART	Adjusted Reference Temperature
ASME	American Society of Mechanical Engineers
ASME Code	ASME Boiler and Vessel Pressure Code
AST	Alternative Source Term
ATWS	Anticipated Transient Without Scram
ATWSI	Anticipated Transient Without Scram with Instability
B-10	Boron-10
BOP	Balance of Plant
BSP	Backup Stability Protection
BTP	Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling-Water Reactor
BWROG	Boiling-Water Reactors Owners Group
CAP	Containment Accident Pressure
CCF	Common-Cause Failure
CDA	Confirmation Density Algorithm
CDF	Core Damage Frequency
CF	Core Flow
CFR	*Code of Federal Regulations*
CHF	Critical Heat Flux
CLTP	Current Licensed Thermal Power
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident

ACRONYM	DEFINITION
CS	Core Spray
CST	Condensate Storage Tank
D3	Defense-in-Depth and Diversity
DBA	Design-Basis Accident
DC	Direct Current
DEGB	Double-Ended Guillotine Break
DSS-CD	Detect and Suppress Solution - Confirmation Density
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFPY	Effective Full Power Years
EOC	End-of-Cycle
EOPs	Emergency Operating Procedures
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
ESF	Engineered Safety Features
ESW	Essential Service Water
EQ	Environmental Qualification
°F	Fahrenheit
FAC	Flow-Accelerated Corrosion
FCV	Flow Control Valve
FFWTR	Final Feedwater Temperature Reduction
FHA	Fuel-Handling Accident
FIV	Flow-Induced Vibration
FMCPR	Final Minimum Critical Power Ratio
FW	Feedwater
FWCF	Feedwater Controller Failure
FWHOOS	Feedwater Heater Out-of-Service
Gd	Gadolinium
GDC	General Design Criterion/Criteria
GE	General Electric
GEH	GE-Hitachi Nuclear Energy
GGNS	Grand Gulf Nuclear Station
GL	Generic Letter
GNF	Global Nuclear Fuel
gpm	gallons per minute
GRA	Growth Rate Algorithm
HCTL	Heat Capacity Temperature Limit
HELB	High Energy Line Break
HPCI	High Pressure Coolant Injection
HPCIL8	Inadvertent HPCI start with Level 8 trip
HRA	Human Reliability Analysis
HSBW	Hot Shutdown Boron Weight

ACRONYM	DEFINITION
HVAC	Heating, Ventilating, and Air Conditioning
I&C	Instrumentation and Control
IASCC	Irradiation Assisted Stress-Corrosion Cracking
ICF	Increased Core Flow
ICPR	Initial Critical Power Ratio
IMCPR	Initial Minimum Critical Power Ratio
kW/ft	kilowatts per foot
LCO	Limiting Condition for Operation
LAR	License Amendment Request
LBLOCA	Large-Break Loss-of-Coolant Accident
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
LTA	Lead Test Assemblies
LRNBP	Load Rejection With No Bypass
LTR	Licensing Topical Report
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MASR	Minimum Alternating Stress Ratio
MCO	Moisture Carry-Over
MCPR	Minimum Critical Power Ratio
M+	(Short for MELLLA+)
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
Mlbm/hr	million pounds mass per hour
MOC	Middle-of-Cycle
MOV	Motor-Operated Valve
MPS	Minimum Recirculation Pump Speed
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVF	Main Steam Isolation Valve Closure with High Flux Scram
MSL	Main Steam Line
MSLB	Main Steam Line Break
MWt	Megawatts Thermal
N-16	Nitrogen-16
NMP2	Nine Mile Point Unit 2
NMS	Neutron Monitoring System

ACRONYM	DEFINITION
NPSH	Net Positive Suction Head
NPSHA	Net Positive Suction Head Available
NRC	U.S. Nuclear Regulatory Commission
NSDC	Normal Shutdown Cooling
NSHC	No Significant Hazards Consideration
NUMAC	Nuclear Measurement Analysis and Control
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OPRM	Oscillation Power Range Monitor
ORNL	Oak Ridge National Laboratory
P	Power
Pa	Containment Leakage Testing Pressure
PBAPS	Peach Bottom Atomic Power Station
PBDA	Period-Based Detection Algorithm
PCT	Peak Cladding Temperature
PRA	Probabilistic Risk Analysis
PRFO	Pressure Regulator Failure Open
PRNM	Power Range Neutron Monitoring
psi	Pounds per Square Inch
psia	Pounds per Square Inch Atmospheric
psig	Pounds per Square Inch Gauge
PTLR	Pressure-Temperature Limits Report
RAI	Request for Additional Information
RBM	Rod Block Monitor
RCF	Rated Core Flow
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
rem	Roentgen Equivalent Man
RFOL	Renewed Facility Operating License
RG	Regulatory Guide
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference
RMS	Root Mean Squared
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
RS	Review Standard
RSD	Replacement Steam Dryer
RSLB	Recirculation Suction Line Break
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup

ACRONYM	DEFINITION
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
S_{AD}	Amplitude Discriminator Setpoint
SAFDL	Specified Acceptable Fuel Design Limit
SAG	Severe Accident Guidelines
SAR	Safety Analysis Report
SBGTS	Standby Gas Treatment System
SBO	Station Blackout
SE	Safety Evaluation
SER	Safety Evaluation Report
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SFP	Spent Fuel Pool
SLO	Single Loop Operation
SORV	Stuck Open Relief Valve
SR	Surveillance Requirement
SRLR	Supplemental Reload Licensing Report
SRO	Strong Rod Out
SRP	Standard Review Plan
SRV	Safety Relief Valve
SRVOOS	Safety relief Valve Out-of-service
SSCs	Structures, Systems, and Components
SSV	Spring Safety Valve
SSW	Sacrificial Shield Wall
STS	Standard Technical Specifications
T-M	Thermal Mechanical
TAF	Top of Active Fuel
TCA	Time Critical Action
THN	Homogeneous Nucleation Temperature
TIP	Traversing Incore Probes
TLO	Two Loop Operation
Tmin	Minimum Temperature for Stable Film Boiling
TS	Technical Specification
TTNBP	Turbine Trip Without Bypass
TTWBP	Turbine Trip With Bypass
UFSAR	Updated Final Safety Analysis Report
USE	Upper Shelf Energy
V&V	Verification and Validation
WRNM	Wide Range Neutron Monitoring
Zr	Zirconium

The NRC staff has determined that its safety evaluation (SE) for the subject amendments contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390. Accordingly, the NRC staff has prepared a redacted, publicly available, non-proprietary version of the SE. Both versions of the SE are enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 305 to Renewed DPR-44
2. Amendment No. 309 to Renewed DPR-56
3. Non-Proprietary Safety Evaluation
4. Proprietary Safety Evaluation

cc w/enclosures 1, 2, and 3: Distribution via Listserv

ADAMS Accession Nos.: ML16034A372 (Cover Letter and Enclosures 1, 2, and 3)
Package: ML16034A396 Enclosure 4 (Proprietary SE): ML16011A442 *by e-mail

OFFICE	DORL/LPL1-2/PM	DORL/LPL1-2/LA	DSS/SRXB/BC	DSS/SCVB/BC	DSS/STSB/BC
NAME	REnnis	LRonewicz	EOesterle	RDennig	RElliott
DATE	2/10/16	2/9/16	2/23/16	2/17/16	2/19/16
OFFICE	DE/EICB/BC	DE/EMCB/BC(A)	DRA/ARCB/BC	DRA/APHB/BC	DRA/APLA/BC*
NAME	MWaters	YLi	UShoop	SWeerakkody	SRosenberg
DATE	2/24/16	2/19/16	2/17/16	2/18/16	2/12/16
OFFICE	OGC	DORL/LPL1-2/BC	DORL/LPL1-2/PM		
NAME	JWachutka	DBroaddus	REnnis		
DATE	3/4/16	3/18/16	3/21/16		

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Letter to B. Hanson from R. Ennis dated March 21, 2016

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS RE: MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS (CAC NOS. MF4760 AND MF4761)

DISTRIBUTION:
PUBLIC
LPL1-2 R/F
RidsNrrDorl Resource
RidsNrrDorlDpr Resource
RidsNrrDorlLpl1-2 Resource
RidsNrrPMPeachBottom Resource
RidsNrrLALRonewicz Resource
RidsRgn1MailCenter Resource
RidsACRS_MailCTR Resource
RidsNrrDss Resource
RidsNrrDssSrxb Resource
RidsNrrDssScvb Resource
RidsNrrDssStsb Resource
RidsNrrDe Resource
RidsNrrDeEmcb Resource
RidsNrrDeEicb Resource
RidsNrrDra Resource
RidsNrrDraApla Resource
RidsNrrDraAphb Resource
RidsNrrDraArcb Resource
RecordsAmend
DSaenz, NRR
GThomas, NRR
MKeefe, NRR
BGreen, NRR
RPedersen, NRR
JDozier, NRR
SPeng, NRR
EEagle, NRR
CBasavaraju, NRR
CJFong, NRR
MChernoff, NRR