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

November 15, 2017

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: Measurement Uncertainty Recapture Power Uprate (CAC Nos. MF9289 and MF9290; EPID L-2017-LLS-0001)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment Nos. 316 and 319 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station, Units 2 and 3. These amendments consist of changes to the Technical Specifications and Renewed Facility Operating Licenses in response to your application dated February 17, 2017,[1] as supplemented by additional letters.[2]

The amendments revise the Renewed Facility Operating Licenses and Technical Specifications to implement a measurement uncertainty recapture power uprate. Specifically, the amendments authorize an increase in the maximum licensed thermal power level from 3,951 megawatts thermal to 4,016 megawatts thermal, which is an increase of approximately 1.66 percent.

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

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML17048A444
[2] Letters dated March 20, 2017 (ADAMS Accession No. ML17080A067); July 13, 2017 (ADAMS Accession No. ML17195A285); August 8, 2017 (ADAMS Accession No. ML17220A214); August 30, 2017 (ADAMS Accession No. ML17243A011); and September 15, 2017 (ADAMS Accession No. ML17258A179)

The NRC staff has determined that its safety evaluation (SE) for these 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
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 316 to Renewed DPR-44
2. Amendment No. 319 to Renewed DPR-56
3. Non-Proprietary SE
4. Proprietary SE

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. 316
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 February 17, 2017, as supplemented by letters dated March 20, 2017; July 13, 2017; August 8, 2017; August 30, 2017; and September 15, 2017, 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.

Enclosure 1

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C(1) and 2.C(2) of Renewed Facility Operating License No. DPR-44 are hereby amended to read as follows:

   (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 4016 megawatts thermal.

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 316, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: November 15, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 316

RENEWED FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| 3 | 3 |

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 |
|---|---|
| 1.1-5 | 1.1-5 |
| 2.0-1 | 2.0-1 |
| 3.2-1 | 3.2-1 |
| 3.2-2 | 3.2-2 |
| 3.2-4 | 3.2-4 |
| 3.3-2 | 3.3-2 |
| 3.3-3 | 3.3-3 |
| 3.3-3a | 3.3-3a |
| 3.3-6 | 3.3-6 |
| 3.3-7 | 3.3-7 |
| 3.3-8 | 3.3-8 |
| 3.3-22 | 3.3-22 |
| 3.3-31a | 3.3-31a |
| 3.3-31b | 3.3-31b |
| 3.3-31c | 3.3-31c |
| 3.4-7 | 3.4-7 |
| 3.5-6 | 3.5-6 |
| 3.5-13 | 3.5-13 |
| 3.7-12 | 3.7-12 |

(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 4016 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 316, 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.

1.1 Definitions

| | |
|---|---|
| PHYSICS TESTS (continued) | b. Authorized under the provisions of 10 CFR 50.59; or<br>c. Otherwise approved by the Nuclear Regulatory Commission. |
| PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) | The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7. |
| RATED THERMAL POWER (RTP) | RTP shall be a total reactor core heat transfer rate to the reactor coolant of 4016 MWt. |
| REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME | The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact up to and including the opening of the trip actuator contacts. |
| RECENTLY IRRADIATED FUEL | RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours. |
| SHUTDOWN MARGIN (SDM) | SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:<br>a. The reactor is xenon free;<br>b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and<br>c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM. |

(continued)

# 2.0 SAFETY LIMITS (SLs)

## 2.1 SLs

### 2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 22.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

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

### 2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

## 2.2 SL Violations

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Any APLHGR not within limits. | A.1 Restore APLHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ 22.6% RTP<br><br><u>AND</u><br><br>In accordance with the Surveillance Frequency Control Program. |

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Any MCPR not within limits. | A.1 Restore MCPR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ 22.6% RTP<br><br>AND<br><br>In accordance with the Surveillance Frequency Control Program. |

(continued)

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Any LHGR not within limits. | A.1 | Restore LHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.3.1 | Verify all LHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ 22.6% RTP<br><br><u>AND</u><br><br>In accordance with the Surveillance Frequency Control Program. |

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| C. One or more automatic Functions with RPS trip capability not maintained.<br><br>OR<br><br>Two or more manual Functions with RPS trip capability not maintained. | C.1 | Restore RPS trip capability. | 1 hour |
| D. Required Action and associated Completion Time of Condition A, B, or C not met. | D.1 | Enter the Condition referenced in Table 3.3.1.1-1 for the channel. | Immediately |
| E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1. | E.1 | Reduce THERMAL POWER to < 26.3% RTP. | 4 hours |
| F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1. | F.1 | Be in MODE 2. | 6 hours |
| G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1. | G.1 | Be in MODE 3. | 12 hours |

(continued)

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 < 17.6% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

-----------------------------------------------------------------------------

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.1.1 | Perform CHANNEL CHECK. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.1.1.2 | ------------------NOTE------------------<br>Not required to be performed until 12 hours after THERMAL POWER ≥ 22.6% RTP.<br>----------------------------------------<br><br>Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 22.6% RTP. | In accordance with the Surveillance Frequency Control Program. |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.1.12 | -------------------NOTES-------------------<br>1. Neutron detectors are excluded.<br><br>2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.<br><br>3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included.<br>-----------------------------------------------<br><br>Perform CHANNEL CALIBRATION. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.1.1.13 | Verify Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions are not bypassed when THERMAL POWER is ≥ 26.3% RTP. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.1.1.14 | Perform CHANNEL FUNCTIONAL TEST. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.1.1.15 | Perform CHANNEL CALIBRATION. | In accordance with the Surveillance Frequency Control Program. |

(continued)

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<br>SR 3.3.1.1.5<br>SR 3.3.1.1.12<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 13 seconds |
| | 5(a) | 3 | H | SR 3.3.1.1.1<br>SR 3.3.1.1.6<br>SR 3.3.1.1.12<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 13 seconds |
| b. Inop | 2 | 3 | G | SR 3.3.1.1.5<br>SR 3.3.1.1.17 | NA |
| | 5(a) | 3 | H | SR 3.3.1.1.6<br>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<br>SR 3.3.1.1.8<br>SR 3.3.1.1.11<br>SR 3.3.1.1.12 | ≤ 15.0% RTP |
| b. Simulated Thermal Power-High | 1 | 3(c) | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.8<br>SR 3.3.1.1.11<br>SR 3.3.1.1.12(e),(f) | ≤ 0.60 W + 65.9% RTP(b)(g) and ≤ 118.0% RTP |
| c. Neutron Flux-High | 1 | 3(c) | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.8<br>SR 3.3.1.1.11<br>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<br>SR 3.3.1.1.11<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | NA |
| f. OPRM Upscale | ≥ 17.6%(h) RTP | 3(c) | I | SR 3.3.1.1.1<br>SR 3.3.1.1.8<br>SR 3.3.1.1.11<br>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) 0.54 (W - ΔW) + 60.3% 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 2 of 3)
Reactor Protection System Instrumentation

| | FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION D.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|
| 3. | Reactor Pressure -High | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 1085.0 psig |
| 4. | Reactor Vessel Water Level-Low (Level 3) | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 1.0 inches |
| 5. | Main Steam Isolation Valve -Closure | 1 | 8 | F | SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 10% closed |
| 6. | Drywell Pressure - High | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 2.0 psig |
| 7. | Scram Discharge Volume Water Level -High | 1,2 | 2 | G | SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 50.0 gallons |
| | | 5(a) | 2 | H | SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17 | ≤ 50.0 gallons |
| 8. | Turbine Stop Valve -Closure | ≥ 26.3% RTP | 4 | E | SR 3.3.1.1.9<br>SR 3.3.1.1.13<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 10% closed |
| 9. | Turbine Control Valve Fast Closure, Trip Oil Pressure -Low | ≥ 26.3% RTP | 2 | E | SR 3.3.1.1.9<br>SR 3.3.1.1.13<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 500.0 psig |
| 10. | Turbine Condenser -Low Vacuum | 1 | 2 | F | SR 3.3.1.1.1<br>SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 21.5 inches Hg vacuum |
| 11. | Deleted | | | | | |
| 12. | Reactor Mode Switch - Shutdown Position | 1,2 | 1 | G | SR 3.3.1.1.14<br>SR 3.3.1.1.17 | NA |
| | | 5(a) | 1 | H | SR 3.3.1.1.14<br>SR 3.3.1.1.17 | NA |

(continued)

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

## 3.3 INSTRUMENTATION

### 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Two channels per trip system of the Digital Feedwater Control System (DFCS) high water level trip instrumentation Function shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

---

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more DFCS high water level trip channels inoperable. | A.1 Place channel in trip. | 72 hours |
| B. DFCS high water level trip capability not maintained. | B.1 Restore DFCS high water level trip capability. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 --------NOTE-------- Only applicable if inoperable channel is the result of inoperable feedwater pump turbine or main turbine stop valve. -------------------- | |
| | Remove affected feedwater pump(s) and main turbine valve(s) from service. | 4 hours |
| | <u>OR</u> | |
| | C.2 Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

## 3.3 INSTRUMENTATION

### 3.3.4.2 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.2

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV)–Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure–Low.

OR

b. The following limits are made applicable:

1. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for inoperable EOC-RPT as specified in the COLR;
2. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR; and
3. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for inoperable EOC-RPT as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 26.3% RTP.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.
-----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more required channels inoperable. | A.1 | Restore channel to OPERABLE status. | 72 hours |
| | OR | | |
| | A.2 | --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- | |
| | | Place channel in trip. | 72 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. One or more Functions with EOC-RPT trip capability not maintained. | B.1 | Restore EOC-RPT trip capability. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 | ---------NOTE-------- Only applicable if inoperable channel is the result of an inoperable RPT breaker. --------------------- | |
| | | Remove the affected recirculation pump from service. | 4 hours |
| | <u>OR</u> | | |
| | C.2 | Reduce THERMAL POWER to < 26.3% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.
----------------------------------------------------------------------------

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.4.2.1 | Perform CHANNEL FUNCTIONAL TEST. | In accordance with the Surveillance Frequency Control Program. |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.4.2.2 | Perform CHANNEL CALIBRATION. The Allowable Values shall be:<br><br>TSV-Closure: ≤ 10% closed; and<br><br>TCV Fast Closure, Trip Oil Pressure-Low: ≥ 500 psig. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.4.2.3 | Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.4.2.4 | Verify TSV-Closure and TCV Fast Closure, Trip Oil Pressure-Low Functions are not bypassed when THERMAL POWER is ≥ 26.3% RTP. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.4.2.5 | -----------------NOTE-----------------<br>Breaker interruption time may be assumed from the most recent performance of SR 3.3.4.2.6.<br>--------------------------------------<br><br>Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.4.2.6 | Determine RPT breaker interruption time. | In accordance with the Surveillance Frequency Control Program. |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.2.1 | -------------------NOTES-------------------<br>1. Not required to be performed until 4 hours after associated recirculation loop is in operation.<br><br>2. Not required to be performed until 24 hours after > 22.6% RTP.<br>------------------------------------------- | |
| | Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop:<br><br>a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns.<br><br>b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.<br><br>c. Each jet pump flow differs by ≤ 10% from established patterns. | In accordance with the Surveillance Frequency Control Program. |

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.8 | -------------------NOTE-------------------<br>Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.<br>------------------------------------------<br><br>Verify, with reactor pressure ≤ 1053 and ≥ 910 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.1.9 | -------------------NOTE-------------------<br>Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.<br>------------------------------------------<br><br>Verify, with reactor pressure ≤ 175 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.1.10 | -------------------NOTE-------------------<br>Vessel injection/spray may be excluded.<br>------------------------------------------<br><br>Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal. | In accordance with the Surveillance Frequency Control Program. |

(continued)

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.3.1 | Verify the RCIC System locations susceptible to gas accumulation are sufficiently filled with water. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.3.2 | -------------------NOTE-------------------- Not required to be met for system vent flow paths opened under administrative control. ------------------------------------------- Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.3.3 | -------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ------------------------------------------- Verify, with reactor pressure ≤ 1053 psig and ≥ 910 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.3.4 | -------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ------------------------------------------- Verify, with reactor pressure ≤ 175 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure. | In accordance with the Surveillance Frequency Control Program. |

(continued)

## 3.7 PLANT SYSTEMS

### 3.7.6 Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 Satisfy the requirements of the LCO. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

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. 319
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 February 17, 2017; as supplemented by letters dated March 20, 2017; July 13, 2017; August 8, 2017; August 30, 2017; and September 15, 2017, 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.

Enclosure 2

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C(1) and 2.C(2) of Renewed Facility Operating License No. DPR-56 are hereby amended to read as follows:

   (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 4016 megawatts thermal.

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 319, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: November 15, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 319

RENEWED FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| 3 | 3 |

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 |
|---|---|
| 1.1-5 | 1.1-5 |
| 2.0-1 | 2.0-1 |
| 3.2-1 | 3.2-1 |
| 3.2-2 | 3.2-2 |
| 3.2-4 | 3.2-4 |
| 3.3-2 | 3.3-2 |
| 3.3-3 | 3.3-3 |
| 3.3-3a | 3.3-3a |
| 3.3-6 | 3.3-6 |
| 3.3-7 | 3.3-7 |
| 3.3-8 | 3.3-8 |
| 3.3-22 | 3.3-22 |
| 3.3-31a | 3.3-31a |
| 3.3-31b | 3.3-31b |
| 3.3-31c | 3.3-31c |
| 3.4-7 | 3.4-7 |
| 3.5-6 | 3.5-6 |
| 3.5-13 | 3.5-13 |
| 3.7-12 | 3.7-12 |

(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 4016 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 319, 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.

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

1.1 Definitions

| | |
|---|---|
| PHYSICS TESTS (continued) | b. Authorized under the provisions of 10 CFR 50.59; or<br><br>c. Otherwise approved by the Nuclear Regulatory Commission. |
| PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) | The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7. |
| RATED THERMAL POWER (RTP) | RTP shall be a total reactor core heat transfer rate to the reactor coolant of 4016 MWt. |
| REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME | The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact up to and including the opening of the trip actuator contacts. |
| RECENTLY IRRADIATED FUEL | RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 312 hours. This 312-hour time period may be reduced to 24 hours if secondary containment hatches H2, H21, H22 and H34 are closed. |
| SHUTDOWN MARGIN (SDM) | SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:<br><br>a. The reactor is xenon free;<br><br>b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and<br><br>c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM. |

(continued)

# 2.0 SAFETY LIMITS (SLs)

## 2.1 SLs

### 2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 22.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

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

### 2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

## 2.2 SL Violations

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Any APLHGR not within limits. | A.1 Restore APLHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ 22.6% RTP<br><br>AND<br><br>In accordance with the Surveillance Frequency Control Program. |

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Any MCPR not within limits. | A.1 Restore MCPR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ 22.6% RTP<br><br>AND<br><br>In accordance with the Surveillance Frequency Control Program. |

(continued)

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Any LHGR not within limits. | A.1 Restore LHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ 22.6% RTP<br><br>AND<br><br>In accordance with the Surveillance Frequency Control Program. |

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. One or more automatic Functions with RPS trip capability not maintained.<br><br><u>OR</u><br><br>Two or more manual Functions with RPS trip capability not maintained. | C.1 Restore RPS trip capability. | 1 hour |
| D. Required Action and associated Completion Time of Condition A, B, or C not met. | D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel. | Immediately |
| E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1. | E.1 Reduce THERMAL POWER to < 26.3% RTP. | 4 hours |
| F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1. | F.1 Be in MODE 2. | 6 hours |
| G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1. | G.1 Be in MODE 3. | 12 hours |

(continued)

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 |
| | <u>AND</u> | | |
| | I.2 | Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power-High scram setpoints defined in the COLR. | 12 hours |
| | <u>AND</u> | | |
| | I.3 | Initiate action 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 |
| | <u>AND</u> | | |
| | J.2 | Reduce operation to below the BSP Boundary defined in the COLR. | 12 hours |
| | <u>AND</u> | | |
| | J.3 | ------ NOTE ---------- LCO 3.0.4 is not applicable. --------------------- Restore required channel to OPERABLE. | 120 days |
| K. Required Action and associated Completion Time of Condition J not met. | K.1 | Reduce THERMAL POWER to < 17.6% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

-----------------------------------------------------------------------------

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.1.1 | Perform CHANNEL CHECK. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.1.1.2 | ------------------NOTE------------------<br>Not required to be performed until 12 hours after THERMAL POWER ≥ 22.6% RTP.<br>----------------------------------------<br><br>Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 22.6% RTP. | In accordance with the Surveillance Frequency Control Program. |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.1.1.12 | -------------------NOTES------------------<br>1. Neutron detectors are excluded.<br>2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.<br>3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included.<br>------------------------------------------<br>Perform CHANNEL CALIBRATION. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.1.1.13 | Verify Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions are not bypassed when THERMAL POWER is ≥ 26.3% RTP. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.1.1.14 | Perform CHANNEL FUNCTIONAL TEST. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.1.1.15 | Perform CHANNEL CALIBRATION. | In accordance with the Surveillance Frequency Control Program. |

(continued)

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<br>SR 3.3.1.1.5<br>SR 3.3.1.1.12<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 13 seconds |
| | 5[(a)] | 3 | H | SR 3.3.1.1.1<br>SR 3.3.1.1.6<br>SR 3.3.1.1.12<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 13 seconds |
| b. Inop | 2 | 3 | G | SR 3.3.1.1.5<br>SR 3.3.1.1.17 | NA |
| | 5[(a)] | 3 | H | SR 3.3.1.1.6<br>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<br>SR 3.3.1.1.8<br>SR 3.3.1.1.11<br>SR 3.3.1.1.12 | ≤ 15.0% RTP |
| b. Simulated Thermal Power-High | 1 | 3[(c)] | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.8<br>SR 3.3.1.1.11<br>SR 3.3.1.1.12[(e),(f)] | ≤ 0.60 W + 65.9% RTP[(b)(g)] and ≤ 118.0% RTP |
| c. Neutron Flux-High | 1 | 3[(c)] | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.8<br>SR 3.3.1.1.11<br>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<br>SR 3.3.1.1.11<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | NA |
| f. OPRM Upscale | ≥ 17.6%[(h)] RTP | 3[(c)] | I | SR 3.3.1.1.1<br>SR 3.3.1.1.8<br>SR 3.3.1.1.11<br>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) 0.54 (W - ΔW) + 60.3% 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 2 of 3)
Reactor Protection System Instrumentation

| | FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION D.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|
| 3. | Reactor Pressure – High | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 1085.0 psig |
| 4. | Reactor Vessel Water Level – Low (Level 3) | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 1.0 inches |
| 5. | Main Steam Isolation Valve – Closure | 1 | 8 | F | SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 10% closed |
| 6. | Drywell Pressure – High | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 2.0 psig |
| 7. | Scram Discharge Volume Water Level – High | 1,2 | 2 | G | SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 50.0 gallons |
| | | 5[(a)] | 2 | H | SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17 | ≤ 50.0 gallons |
| 8. | Turbine Stop Valve – Closure | ≥ 26.3% RTP | 4 | E | SR 3.3.1.1.9<br>SR 3.3.1.1.13<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≤ 10% closed |
| 9. | Turbine Control Valve Fast Closure, Trip Oil Pressure – Low | ≥ 26.3% RTP | 2 | E | SR 3.3.1.1.9<br>SR 3.3.1.1.13<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 500.0 psig |
| 10. | Turbine Condenser – Low Vacuum | 1 | 2 | F | SR 3.3.1.1.1<br>SR 3.3.1.1.9<br>SR 3.3.1.1.15<br>SR 3.3.1.1.17<br>SR 3.3.1.1.18 | ≥ 21.5 inches Hg vacuum |
| 11. | Deleted | | | | | |
| 12. | Reactor Mode Switch – Shutdown Position | 1,2 | 1 | G | SR 3.3.1.1.14<br>SR 3.3.1.1.17 | NA |
| | | 5[(a)] | 1 | H | SR 3.3.1.1.14<br>SR 3.3.1.1.17 | NA |

(continued)

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

## 3.3 INSTRUMENTATION

### 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Two channels per trip system of the Digital Feedwater Control System (DFCS) high water level trip instrumentation Function shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.
----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more DFCS high water level trip channels inoperable. | A.1 | Place channel in trip. | 72 hours |
| B. DFCS high water level trip capability not maintained. | B.1 | Restore DFCS high water level trip capability. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 | --------NOTE--------- Only applicable if inoperable channel is the result of inoperable feedwater pump turbine or main turbine stop valve. -------------------- | |
| | | Remove affected feedwater pump(s) and main turbine valve(s) from service. | 4 hours |
| | OR | | |
| | C.2 | Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

## 3.3 INSTRUMENTATION

### 3.3.4.2 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.2

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV)–Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure–Low.

OR

b. The following limits are made applicable:

1. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for inoperable EOC-RPT as specified in the COLR;
2. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR; and
3. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for inoperable EOC-RPT as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 26.3% RTP.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.
----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more required channels inoperable. | A.1 | Restore channel to OPERABLE status. | 72 hours |
| | OR | | |
| | A.2 | --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- | |
| | | Place channel in trip. | 72 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. One or more Functions with EOC-RPT trip capability not maintained. | B.1 Restore EOC-RPT trip capability. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 --------NOTE--------- Only applicable if inoperable channel is the result of an inoperable RPT breaker. -------------------- Remove the affected recirculation pump from service. | 4 hours |
| | OR | |
| | C.2 Reduce THERMAL POWER to < 26.3% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.
----------------------------------------------------------------------------

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.3.4.2.1 Perform CHANNEL FUNCTIONAL TEST. | In accordance with the Surveillance Frequency Control Program. |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.4.2.2 | Perform CHANNEL CALIBRATION. The Allowable Values shall be:<br><br>TSV-Closure: ≤ 10% closed; and<br><br>TCV Fast Closure, Trip Oil Pressure-Low: ≥ 500 psig. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.4.2.3 | Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.4.2.4 | Verify TSV-Closure and TCV Fast Closure, Trip Oil Pressure-Low Functions are not bypassed when THERMAL POWER is ≥ 26.3% RTP. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.4.2.5 | -------------------NOTE-------------------<br>Breaker interruption time may be assumed from the most recent performance of SR 3.3.4.2.6.<br>------------------------------------------<br><br>Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.3.4.2.6 | Determine RPT breaker interruption time. | In accordance with the Surveillance Frequency Control Program. |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.2.1 | -------------------NOTES-------------------<br>1. Not required to be performed until 4 hours after associated recirculation loop is in operation.<br><br>2. Not required to be performed until 24 hours after > 22.6% RTP.<br>-------------------------------------------<br><br>Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop:<br><br>a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns.<br><br>b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.<br><br>c. Each jet pump flow differs by ≤ 10% from established patterns. | In accordance with the Surveillance Frequency Control Program. |

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.8 | -------------------NOTE-------------------<br>Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.<br>------------------------------------------<br><br>Verify, with reactor pressure ≤ 1053 and ≥ 910 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.1.9 | -------------------NOTE-------------------<br>Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.<br>------------------------------------------<br><br>Verify, with reactor pressure ≤ 175 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.1.10 | -------------------NOTE-------------------<br>Vessel injection/spray may be excluded.<br>------------------------------------------<br><br>Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal. | In accordance with the Surveillance Frequency Control Program. |

(continued)

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.3.1 | Verify the RCIC System locations susceptible to gas accumulation are sufficiently filled with water. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.3.2 | ------------------NOTE------------------<br>Not required to be met for system vent flow paths opened under administrative control.<br>----------------------------------------<br><br>Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.3.3 | ------------------NOTE------------------<br>Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.<br>----------------------------------------<br><br>Verify, with reactor pressure ≤ 1053 psig and ≥ 910 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.5.3.4 | ------------------NOTE------------------<br>Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.<br>----------------------------------------<br><br>Verify, with reactor pressure ≤ 175 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure. | In accordance with the Surveillance Frequency Control Program. |

(continued)

## 3.7 PLANT SYSTEMS

### 3.7.6 Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 22.6% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 Satisfy the requirements of the LCO. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 22.6% RTP. | 4 hours |

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 316 AND 319

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 bolded double brackets
as shown here **[[ ]]**.

Enclosure 3

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**APPENDICES**

## 1.0 **INTRODUCTION**

By application dated February 17, 2017 (Reference 1), as supplemented by letters dated March 20, 2017 (Reference 2); July 13, 2017 (Reference 3); August 8, 2017 (Reference 12); August 30, 2017 (Reference 39); and September 15, 2017 (Reference 44), 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 amendments would revise the Renewed Facility Operating Licenses (RFOLs) and Technical Specifications (TSs) to implement a measurement uncertainty recapture (MUR) power uprate. Specifically, the amendments would authorize an increase in the maximum licensed thermal power level from 3,951 megawatts thermal (MWt) to 4,016 MWt, which is an increase of approximately 1.66 percent.

The supplemental letters dated March 20, 2017; July 13, 2017; August 8, 2017, August 30, 2017; and September 15, 2017, 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 or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 2, 2017 (82 FR 20497).

### 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 for PBAPS*

The Atomic Energy Commission (AEC) 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 3,293 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 3,293 MWt to 3,458 MWt.

By Amendment Nos. 247 and 250 (Units 2 and 3, respectively), dated November 22, 2002, the NRC approved a 1.62 percent MUR power uprate that authorized an increase in the maximum thermal power level from 3,458 MWt to 3,514 MWt.

By Amendment Nos. 293 and 296 (Units 2 and 3, respectively) dated August 25, 2014 (Reference 4), the NRC approved a 12.4 percent extended power uprate (EPU) that authorized an increase in the maximum thermal power level from 3,514 MWt to the current licensed thermal power (CLTP) level of 3,951 MWt. The EPU power level of 3,951 MWt represents an increase of approximately 20 percent above the OLTP level of 3,293 MWt.

In addition to the power uprates noted above, by Amendment Nos. 305 and 309 (Units 2 and 3, respectively), dated March 21, 2016 (Reference 7), the NRC approved a change to the PBAPS TSs and RFOLs to allow plant operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain. The 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.

*General Approach for MUR Power Uprates*

Nuclear power plants are licensed to operate at a specified maximum core thermal power often called rated thermal power (RTP). Appendix K, "[Emergency Core Cooling System] ECCS Evaluation Models," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, formerly required licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and ECCS analyses. This requirement was included to ensure that instrumentation uncertainties were adequately accounted for in the safety analyses. In practice, many of the design-bases analyses assumed a 2 percent power uncertainty, consistent with 10 CFR Part 50, Appendix K.

A change to 10 CFR Part 50, Appendix K, was published in the *Federal Register* on June 1, 2000 (65 FR 34913), which became effective July 31, 2000. This change allows licensees to use a power level less than 1.02 times the RTP for the LOCA and ECCS analyses, but not a power level less than the licensed power level, based on the use of state-of-the art feedwater (FW) flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses, provided that the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. As there continues to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA is preserved. However, this change to 10 CFR 50, Appendix K, did not authorize increases in licensed power levels for individual nuclear power plants. As the licensed power level for a plant is contained in its operating license, licensees seeking to raise the licensed power level must submit an LAR, which must be reviewed and approved by the NRC staff.

In existing nuclear power plants, the neutron flux instrumentation continuously indicates the reactor core thermal power. This instrumentation must be periodically calibrated to accommodate the effects of fuel burnup, flux pattern changes, and instrumentation setpoint drift. The reactor core thermal power generated by a nuclear power plant is determined by steam plant calorimetry, which is the process of performing a heat balance around the nuclear steam supply system (called a calorimetric). The accuracy of this calculation depends primarily upon the accuracy of FW flow rate and FW net enthalpy measurements. As such, an accurate measurement of FW flow rate and temperature is necessary for an accurate calibration of the nuclear instrumentation. Of the two parameters, flow rate and temperature, the most important in terms of calibration sensitivity is the FW flow rate.

The instruments originally installed to measure FW flow rate in existing nuclear power plants were usually a venturi or a flow nozzle, each of which generates a differential pressure proportional to the square of the FW velocity in the pipe. However, error in the determination of flow rate can be introduced due to venturi fouling and, to a lesser extent, flow nozzle fouling, the transmitter, and the analog-to-digital converter. As a result of the desire to reduce flow instrumentation uncertainty to enable operation of the plant at a higher power while remaining bounded by the accident analyses, the industry assessed alternate flow rate measurement techniques and found that ultrasonic flow meters (UFMs) are a viable alternative. UFMs are based on computer-controlled electronic transducers that do not have differential pressure elements that are susceptible to fouling.

### 1.2 Licensee's Approach

As discussed in Section 1.0 of Attachment 1 to the licensee's application, the LAR is based on the increased FW flow measurement accuracy of the Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) UFM instrumentation, which provides a more accurate calculation of reactor thermal power. The CheckPlus™ LEFM system was installed in PBAPS, Unit 2, in 2002 and in PBAPS, Unit 3, in 2003, to support the first MUR power uprate, which was approved by the NRC in November 2002. However, the analyses conducted in support of the EPU approved in August 2014 assumed a 2 percent RTP uncertainty and did not take credit for the increased accuracy provided by the LEFM system.

As discussed in Section 3.2 of Attachment 1 to the licensee's application, with credit for the LEFM system, the core thermal power measurement uncertainty will be a maximum of 0.34 percent. As such, the licensee stated this will support an increase in RTP of 1.66 percent (i.e., 2.00 – 0.34), from 3,951 MWt to 4,016.6 MWt, which is conservatively rounded down to the requested value of 4,016 MWt.

In order to provide guidance to licensees seeking an MUR power uprate on the basis of improved FW flow measurement, the NRC issued Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002 (Reference 5). RIS 2002-03 provides guidance on the scope and detail of the information that should be provided to the NRC staff for MUR power uprate LARs. While RIS 2002-03 does not constitute an NRC requirement, its use aids licensees in the preparation of their MUR power uprate LAR, while also providing guidance to the NRC staff for the conduct of its review. The licensee stated in Section 3.2 of Attachment 1 to its application that the scope and content of the evaluations performed and described in the LAR are consistent with the guidance contained in RIS 2002-03. Attachment 4 to Exelon's application provides a cross-reference between the contents of the LAR and the guidance in RIS 2002-03.

RIS 2002-03 also states that "where the NRC has approved a specific methodology (e.g., topical report) for the type of measurement uncertainty recapture power uprate being requested, licensees should follow the format prescribed for that specific methodology and provide the information called for in that methodology and the NRC's letter and safety evaluation (SE) approving the methodology." By letter and SE dated April 1, 2003, the NRC approved a

General Electric (GE) Licensing Topical Report (LTR) that provides a methodology acceptable to the staff for justifying an MUR power uprate for a BWR. The accepted version of this Thermal Power Optimization (TPO) LTR (referred to as the TLTR), NEDC-32938P-A, Revision 2, was issued in May 2003 (Reference 6).

Attachment 5 to Exelon's application dated February 17, 2017, GE - Hitachi Nuclear Energy (GEH), "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," NEDC-33873P, Revision 0, dated February 2017, summarizes the evaluations performed for PBAPS in accordance with the content and format specified in the TLTR. This proprietary report is referred to as the "TSAR" (Thermal Power Optimization Safety Analysis Report). A public version of the TSAR, GEH report NEDO-33873, is contained in Attachment 7 to Exelon's application. Attachment 14 to Exelon's application contains a redline/strikeout of the GEH TSAR template to show the changes made to generate the PBAPS plant-specific TSAR provided in Attachment 5.

### 1.3 Method of NRC Review

The NRC staff's review evaluated the licensee's assessment of the impact of the proposed MUR on the applicable PBAPS design-basis analyses. The NRC staff reviewed the licensee's application and supplements.

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 MUR conditions.

Details of the NRC staff's technical evaluation are provided in Section 3.0 of this SE. The technical evaluation generally follows the format of the technical review areas contained in Sections 2.0 through 10.0 of the TSAR. As noted above, the TSAR was provided in Attachment 5 (proprietary, non-public) and Attachment 7 (public) to the licensee's application.

## 2.0 REGULATORY EVALUATION

### 2.1 General Design Criteria

The construction permit for PBAPS, Units 2 and 3, was issued by the 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 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 (Agencywide Documents Access and Management System (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 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 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 34, "Reactor Coolant Pressure Boundary rapid Propagation Failure Prevention (Category A)," which requires, in part, that the RCPB be designed to minimize the probability of rapidly propagating failures.

- 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 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.

- Final GDC-3, "Fire protection," which requires, in part, that that structures, systems, and components (SSCs) important to safety be designed and located to minimize the probability and effect of fires, noncombustible and heat resistant materials be used, and fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety.

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

- Final GDC 10, "Reactor design," which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified, acceptable fuel design limits (SAFDLs) 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 14, "Reactor coolant pressure boundary," which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture and of gross rupture.

- Final GDC 17, "Electric power systems," which requires, in part, that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of SSCs important to safety.

- 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 non-brittle manner, and the probability of rapidly propagating fracture is minimized.

- Final GDC 60, "Control of releases of radioactive materials to the environment," which requires, in part, that the plant design include means to control the release of radioactive effluents.

- Final GDC 61, "Fuel storage and handling and radioactivity control," which requires, in part, that systems that contain radioactivity be designed with appropriate confinement.

## 2.2 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) limiting conditions for operation (LCOs); (3) surveillance requirements (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)(1), safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down.

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.

## 2.3 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 50.48, "Fire protection," and 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," insofar as they require the development of a fire protection program to ensure, among other things, the capability to safely shut down the plant.

- 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 ECCS accident performance and acceptance criteria for that calculated performance.

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," which, in part, requires licensees to establish programs to qualify electric equipment important to safety.

- 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," which requires compliance with the fracture toughness and material surveillance program requirements for the RCPB set forth in 10 CFR Part 50, Appendix G and Appendix H.

- 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 50.120, "Training and qualification of nuclear power plant personnel," which requires, in part, that the licensee establish, implement, and maintain a training program.

- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," which provides quality assurance requirements for the design, fabrication, construction, and testing of SSCs.

- 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," provides fracture toughness requirements for ferritic materials in the RCPB, including requirements on the upper-shelf energy (USE) values used for assessing the safety margins of the reactor vessel materials against ductile tearing and requirements for calculating pressure-temperature (P-T) limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including AOOs and hydrostatic tests.

- 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region.

- 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.

### 2.4 Guidance Documents

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

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 13).
- NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2, dated May 2002 (Reference 14).
- NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, dated November 2012 (Reference 15).
- NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, dated September 30, 2007 (Reference 16).
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Reference 17).

## 3.0 TECHNICAL EVALUATION

### 3.1 Overview of the Thermal Power Optimization Safety Analysis Report

As discussed in SE Section 1.2, the TSAR, GEH report NEDC-33873P (Attachment 5 to Exelon's application), summarizes the plant-specific evaluations performed to support the proposed MUR power uprate for PBAPS. The TSAR contains information divided into the following sections:

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

The TSAR also contains Appendices A, B, C, and D, which address continued applicability of the limitations and conditions described in the NRC SEs for four GEH LTRs at MUR RTP conditions, including plant conditions based on the previously approved EPU and MELLLA+ amendments. The limitations and conditions addressed in TSAR Appendices A, B, C, and D apply to the following NRC-approved LTRs, respectively:

- NEDC-33173P-A, Revision 4, "Applicability of GE Methods to Expanded Operating Domains" (Reference 8). This LTR is referred to as the "Methods LTR" throughout this SE.

- NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus" (Reference 9). This LTR is referred to as the "M+ LTR" throughout this SE.
- NEDC-33075P-A, Revision 8, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density [DSS-CD]" (Reference 10). This LTR is referred to as the "DSS-CD LTR" throughout this SE.
- NEDE-32906P, Supplement 3-A, Revision 1, "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients" (Reference 11).

As discussed above in SE Section 1.3, the NRC staff's technical evaluation generally follows the format of the technical review areas contained in Sections 2.0 through 10.0 of the TSAR. These technical review areas are addressed in SE Sections 3.2 through 3.10.

## 3.2 TSAR Section 2.0 – Reactor Core and Fuel Performance

The following provides the NRC staff's technical review of the topics in Section 2.0 of the TSAR.

### 3.2.1 TSAR Section 2.1 – Fuel Design and Operation

The licensee stated in TSAR Section 2.1 that at the MUR RTP conditions, all fuel and core design criteria will be met by the deployment of fuel enrichment and burnable poison, control rod pattern management, and core flow adjustments. The licensee also stated that revised loading patterns, slightly larger batch sizes, and potentially new fuel designs may be used to provide additional operating flexibility and maintain fuel cycle length. In addition, the NRC-approved limits for burnup on the fuel will not be exceeded.

The NRC staff finds that the above evaluation for fuel and core designs at the MUR uprated power level conforms to the acceptance criteria of Section 4.2 of the Standard Review Plan (Reference 17) regarding fuel evaluation, and the guidance provided in RIS 2002-03 regarding analyses that bound plant operation. Therefore, the NRC staff concludes that the evaluation provides reasonable assurance that final GDC 10 will continue to be met under TPO conditions.

*TSAR Section 2.1.1 – Fuel Product Line*

The licensee stated in TSAR Section 2.1.1 that implementation of the MUR does not necessitate a new fuel design and, therefore, no additional fuel and core design evaluations are required. However, to demonstrate the change from the current 100 percent RTP to the 101.66 percent MUR RTP is inconsequential, the licensee performed the same fuel and core design evaluations in the TSAR that were performed for the PBAPS MELLLA+ amendment. The licensee stated that the magnitude of the changes to thermal margins and other core characteristics are within normal cycle-to-cycle variation.

Many of the current safety analyses for PBAPS were performed at 102 percent RTP, which bounds the MUR RTP of 101.66 percent. Therefore, the NRC staff does not anticipate a significant change in the magnitude of thermal margins and core characteristics at the uprated power when compared to the prior analyses. Examination of the fuel and core design evaluations presented in the TSAR in Figures 2-1 through 2-17 confirm this (e.g., a slight increase **[[ ]]** in the limiting bundle operating minimum critical power ratio (MCPR) is observed as expected from the increased core flow at MUR conditions, and minor increases **[[ ]]** in linear heat generation rate (LHGR) for the hottest channels as expected from the increased core power). The changes observed in these evaluations are within the typical range expected for cycle-to-cycle core performance variation. Because the current fuel product line was designed anticipating normal cycle-to-cycle variations, the NRC staff finds the licensee's assessment that implementation of the MUR does not require a new fuel design to be acceptable.

Additionally, the licensee states that the fuel product line design for PBAPS will be evaluated for the reload prior to the MUR implementation, consistent with the fuel product line requirements in NEDC-33270P, Revision 7 (Reference 47). The use of NRC-approved fuel design acceptance criteria and analysis methodologies ensure that the fuel bundles perform in a manner that is consistent with the objectives of Sections 4.2 and 4.3 of the NRC's SRP (Reference 17) and final GDC 10. Therefore, the NRC staff finds this acceptable.

*TSAR Section 2.1.2 – Core Design*

As discussed in TSAR Section 1.2.1, and shown in TSAR Figures 1-1a and 1-1b, the TPO operating domain is established by extending the current MELLLA+ upper boundary with no increase in the maximum core flow. As discussed in TSAR Section 2.1.2, the TPO operating domain allows for higher bundle powers but not lower bundle flows, resulting in core power-to-flow ratios less than at the current RTP. Therefore, the NRC staff anticipates that the range of void fraction, axial and radial power shape, and the magnitude of steady-state bypass boiling will change slightly between pre-MUR and post-MUR power uprate operating conditions.

In the case of bypass voiding, instrumentation-specific design bases limit the presence of bypass voiding to **[[ ]]**. This is documented in the Methods LTR (Reference 8). Limiting bypass voiding ensures that instrumentation is operated within specifications. To demonstrate that the change in bypass voiding satisfies this limit for MELLLA+ and MUR power-uprated conditions, the licensee provided plant-specific steady-state bypass voiding predictions in TSAR Table 2-1. The predictions are given at several core power and flow state points for the hot fuel channel. All of the bypass void fraction predictions are less than the required limitation. Therefore, the NRC staff finds this acceptable.

In the case of void fraction range and axial and radial power shape, these changes may affect the power distribution in the core, which, in turn, may negatively impact fuel thermal design margins. Limitation and Condition 9.24 for the Methods LTR requires that plant-specific applications at EPU and MELLLA+ provide predictions of key parameters for cycle exposures and quarter-core maps so that core conditions and fuel design margins may be assessed. Therefore, the licensee provided predictions of key parameters for each cycle exposure state point of the TSAR core design. The licensee plotted these predictions against several reference plants and the PBAPS MELLLA+ analyses. The key parameter comparisons are presented in TSAR Figures 2-1 through 2-6 and consist of peak bundle power, peak bundle coolant flow, peak bundle exit void fraction, maximum channel exit void fraction, core average void fraction, and peak LHGR. The licensee also provided quarter-core maps in TSAR Figures 2-7 through 2-15 showing bundle power, operating LHGR, and MCPR for beginning-of-cycle (0 megawatt-days per short ton (MWd/ST)), middle-of-cycle (8,000 MWd/ST), and end-of-rated (15,850 MWd/ST).

The licensee indicated that TSAR Figures 2-1 through 2-6 demonstrate that the changes in power distribution are minimal. The NRC staff examined TSAR Figures 2-1 through 2-6 and observed that the peak bundle power and peak bundle coolant flow are in the expected range as compared to the PBAPS MELLLA+ analyses. The staff took note that the exit voiding at PBAPS is higher than the reference plants, but this is to be expected, since PBAPS operates at a higher power density and lower core flow throughout the entire cycle. Likewise, the staff observed that the PBAPS TPO exit voiding is slightly less than that of the PBAPS MELLLA+. This is also to be expected because while the MUR-expanded MELLLA+ domain offers higher power density, it comes with increased core flow. Thus, more voids are swept out of the bundle for a given power, causing a slight decrease in bundle exit void fraction when compared to the prior PBAPS MELLLA+ Safety Analysis Report (SAR) analyses. The staff, therefore, agrees with the licensee's assessment that the change in power distribution is minimal.

The licensee also indicated that the appropriate power distribution in the core will be achieved while maintaining individual fuel bundles within allowable thermal limits. Given the minimal change in power distribution discussed above, the NRC staff does not anticipate a significant change in the magnitude of thermal margins at the uprated power when compared to the prior PBAPS MELLLA+ analyses. The staff examined the quarter-core maps presented in TSAR Figures 2-7 through 2-16 in comparison to the PBAPS MELLLA+ analyses and confirmed this. A slight increase **[[ ]]** in the limiting bundle operating MCPR is observed, as expected, from the increased core flow at MUR conditions. Minor increases **[[ ]]** in LHGR for the hottest channels are also observed, as expected, from the increased core power density. The changes observed in these thermal limits do not stress the fuel thermal design limits. Additionally, the changes are within the expected range for cycle-to-cycle core performance variation. Because the current fuel product line was designed anticipating these normal cycle-to-cycle variations, the NRC staff concludes that the fuel bundles will be maintained within allowable thermal limits.

As discussed above, TSAR Section 2.1 states that at the MUR uprate conditions, all fuel and core design will be met by the deployment of fuel enrichment and burnable poison, control rod pattern management, and core flow adjustment. Also, as discussed above, no new fuel designs are introduced in order to meet the MUR uprate conditions. However, revised loading patterns and slightly larger batch sizes may be used to provide additional operating flexibility and maintain fuel cycle length. The NRC staff finds the approach acceptable because the use of NRC-approved fuel design acceptance criteria and analysis methodologies will ensure that the fuel bundles perform in a manner that is consistent with the objectives of Sections 4.2 and 4.3 of the NRC's SRP and final GDC-10. The licensee will perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design acceptance criteria during steady-state, AOO, and accident conditions.

*TSAR Section 2.1.3 – Fuel Thermal Monitoring Threshold*

As discussed in TSAR Section 2.1.3, the power level above which fuel thermal margin monitoring is required is typically set at 25 percent of RTP. The use of 25 percent RTP is based on analyses applicable to the BWR plant with the **[[ ]]**. As originally licensed, the **[[ ]]** for this plant at 100 percent RTP is **[[ ]]**. Analyses performed for this plant demonstrate that at 25 percent of RTP (**[[ ]]**), a substantial margin exists that is adequate to ensure fuel thermal limits are not exceeded, even in the event of a limiting transient. Because this plant possessed [[ ]], these analyses were bounding for all other plants. Thus, the use of 25 percent RTP was adopted as the fuel thermal margin monitoring threshold.

However, an increase in a plant's RTP, such as the result of a power uprate, will cause an increase in [[ ]]. If the increase in RTP is great enough, it may result in [[ ]] at 25 percent RTP exceeding [[ ]], in which case the historical analyses justifying the use of 25 percent RTP as a monitoring threshold are no longer bounding. Section 5.1.3 of the TLTR SE indicates that, in such a case, the fuel thermal margin monitoring threshold is scaled down to ensure that monitoring is initiated [[ ]].

For PBAPS, the licensee stated in Section 2.1.3 of the TSAR that the thermal limits monitoring threshold stated in the TS Safety Limits, LCOs, and SRs will be reduced from the EPU and MELLLA+ value of 23 percent RTP to 22.6 percent RTP for the MUR power uprate. The NRC staff assessed the new fuel thermal monitoring threshold and found it acceptable because PBAPS' [[ ]] at 22.6 percent of MUR uprated power does not exceed the previously licensed maximum, which is in accordance with Section 5.8 of the TLTR. The choice of power level for the fuel thermal margin monitoring threshold at MUR uprated power meets the acceptance criteria of SRP Section 4.4 regarding ensuring adequate safety margin and provides reasonable assurance that the final GDC 10 criteria will continue to be met. Therefore, the NRC staff finds the fuel thermal limits monitoring threshold to be acceptable.

### 3.2.2 TSAR Section 2.2 – Thermal Limits Assessment

Final GDC 10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified, acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs. Operating thermal limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents). The safety limit minimum critical power ratio (SLMCPR) protects 99.9 percent of the fuel rods from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as result of an AOO. The operating linear heat generation rate (LHGR) is the core operating limit that assures the fuel thermal-mechanical performance limit (i.e., the 1 percent fuel plastic strain design limit or the no-fuel-centerline-melt criterion) will not be exceeded as a result of an AOO. This section addresses the effects of the TPO uprate on the thermal limits.

*SLMCPR*

The SLMCPR is calculated based on the actual core loading pattern for each reload core in accordance with the methods defined in the NRC-approved Global Nuclear Fuel (GNF) Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (Reference 45). 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. The SLMCPR is dependent upon the nominal average power level and the uncertainty in its measurement.

As required by M+ LTR (Reference 9) Limitation and Condition 12.6, the SLMCPR will be calculated at the rated statepoint (100.0% of TPO RTP / 100.0% of core flow (CF)), the upper left corner of the MELLLA+ upper boundary (100% of TPO RTP / 85.2% of CF), the lower left corner of the MELLLA+ upper boundary (77.5% of TPO RTP / 55.0% of CF), and the TPO RTP at the increased core flow (ICF) statepoint (100.0% of TPO RTP / 110.0% of CF) (i.e., see TSAR Figure 1-1a, Statepoints E, J, K, and F, respectively).

As required by Methods LTR (Reference 8) Limitation and Condition 9.5, the cycle-specific SLMCPR determined based on M+ LTR Limitation and Condition 12.6 will also include either a +0.02 SLMCPR adder for operation at statepoints with a power-to-flow ratio greater than 42 megawatts thermal per million pounds mass per hour (MWt/million pounds mass per hour (Mlbm)/hr), or a +0.01 SLMCPR adder for operation at statepoints with a power-to-flow ratio less than 42 MWt/Mlbm/hr. The cycle-specific SLMCPR analysis will incorporate either a +0.01 or a +0.02 SLMCPR adder for MELLLA+ operation, including the expanded TPO region. The calculated values will be documented in the Supplemental Reload Licensing Report (SRLR). Since the SLMCPR values are specified in TS 2.1.1.2, the licensee would be required to submit an LAR if the current values are not bounding. By letter dated September 15, 2017 (Reference 44), the licensee submitted the SRLR for PBAPS, Unit 3, Cycle 22 (i.e., operating cycle following the fall 2017 refueling outage). As shown on page 19 of the SRLR, the SLMCPR values for single loop and two loop operation were determined to be 1.15, which is the same as the values currently shown in TS 2.1.1.2.

*OLMCPR*

The OLMCPR is determined on a cycle-specific basis from the results of the reload transient analysis, and this approach will not change for the TPO uprate. AOOs are analyzed at various points in the allowable operating domain, depending on the type of transient. The change in the MCPR is combined with the SLMCPR to establish the OLMCPR, which ensures that 99.9 percent of the rods will not reach boiling transition in the event of an anticipated transient. The OLMCPR is determined on a cycle-specific basis from the results of the reload transient analysis, as described in GESTAR II. The cycle-specific analysis results are documented in the SRLR and included in the core operating limits report. The MELLLA+ operating conditions, including the expanded TPO region, do not change the methods used to determine this limit.

*Maximum Planar Linear Heat Generation Rate Limits*

The maximum planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the licensee performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and **[[**

**]]**

The reload design process for PBAPS ensures that the MAPLHGR limits will be met for each reload. The MAPLHGR limits ensure that the plant does not exceed regulatory limits established in 10 CFR 50.46 and demonstrate that plants meet the regulatory limits in the MELLLA+ operating domain, including the expanded TPO region. The MAPLHGR limits for PBAPS will be evaluated for the reload core prior to TPO implementation.

*LHGR Limits*

The steady-state and transient LHGR limits are established for every fuel design to protect against fuel centerline melt throughout the operating cycle. The licensee will determine the LHGR limits for the uprated cycle in the reload analysis for future cycles, and these limits will be maintained during operation.

The PBAPS LHGR limits ensure that the plant does not exceed fuel thermal-mechanical design limits. There are no changes to the PBAPS fuel or fuel design limits as a result of MELLLA+ domain expansion with TPO. PBAPS continues to use the GNF2 fuel product line consistent with the GNF2 requirements. The TPO operating conditions do not change the methods used to determine this limit. The LHGR limits for PBAPS will be evaluated for the reload core prior to TPO implementation.

*Power-to-Flow Ratio*

The TPO expanded MELLLA+ operating domain allows for higher bundle powers, but not lower bundle flows, due to the extension of the existing MELLLA+ boundary. The bundle power-to-flow ratios at TPO 122 percent of OLTP core power and 85.2 percent CF conditions are less than the bundle power-to-flow ratios at the previous MELLLA+ 120 percent of OLTP core power and 83.0 percent CF. Therefore, the current analysis of record based on MELLLA+ analysis bounds the TPO operation.

*Thermal Limits Conclusion*

Because the OLMCPR, SLMCPR, LHGR, and MAPLGHR values for the uprated conditions are calculated as part of each reload analysis using NRC-approved methodologies to ensure that SAFDLs are not violated, and appropriate changes to the safety limits are made in the TSs and/or the core operating limits report, the NRC staff concludes that the licensee's assessment of thermal limits for the TPO uprate is acceptable.

3.2.3 <u>TSAR Section 2.3 – Reactivity Characteristics</u>

The reload core analysis will ensure that the minimum shutdown margin requirements will be met for each core design. All minimum shutdown margin requirements apply to cold shutdown conditions and are maintained without change. Checks of cold shutdown margin based on SLC system boron injection capability and shutdown using control rods with the most reactive control rod stuck out are made for each reload. The TPO uprate has no significant effect on these conditions; the shutdown margin is confirmed in the reload core design.

The MELLLA+ operating conditions, including the expanded TPO region, do not change the PBAPS methods used to evaluate that the strongest rod out shutdown margin meets the current PBAPS design and TS cold shutdown margin requirements. The MELLLA+ operating conditions, including the expanded TPO region, do not change the PBAPS methods used to evaluate that SLC system shutdown margin meets the current PBAPS design and the SLC system TS requirements.

Based on the above considerations, the NRC staff concludes that the TPO uprate is acceptable regarding the licensee's assessment of shutdown margin.

3.2.4 TSAR Section 2.4 – Thermal Hydraulic Stability

*Detect and Suppress Solution - Confirmation Density*

As discussed in TSAR Section 2.4 and Section 3.2.2 of the NRC staff's SE for the MELLLA+ amendments (Reference 7), PBAPS is currently operating under the detect and suppress solution - confirmation density (DSS-CD) long-term stability solution, consistent with the DSS-CD LTR (Reference 10), including any applicable limitations and conditions. The DSS-CD stability solution has been shown to provide an early trip signal upon instability inception for both core wide and regional mode oscillations. The DSS-CD solution monitors oscillation power range monitor (OPRM) signals to determine when a reactor scram is required. The OPRM signal is evaluated by the DSS-CD stability algorithms to determine when the signal is becoming sufficiently periodic and large to warrant a reactor scram to disrupt the oscillation.

The licensee stated in TSAR Section 2.4.1 that **[[**

**]]**

The plant-specific application demonstrates that the analyses and evaluations supporting DSS-CD are applicable to the fuel loaded in the core and the new operating power domain. Since the approach described above is based on an NRC-approved methodology, the staff concludes that the licensee's evaluation with respect the DSS-CD stability solution is acceptable for the TPO uprate.

*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.

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 thermal limits monitoring threshold of 25 percent OLTP and rated recirculation drive flow ≤ 75 percent. For a power-uprated plant, the thermal limits monitoring threshold may be scaled to a lower percent value.

Currently, the OPRM Armed Region for PBAPS is defined as the region on the power/flow map with power ≥ 23.0 percent of RTP and rated recirculation drive flow ≤ 75 percent. As discussed in TSAR Section 2.4.3, at TPO conditions, the new OPRM Armed Region power boundary will be rescaled to 22.6 percent of the TPO power level. Since the rated core flow does not change for TPO, the 75 percent core flow boundary is not rescaled.

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 thermal limits monitoring threshold level. The LTR states that the alternate method is acceptable because system operability is assured prior to entry into the Armed Region. Accordingly, for the TPO uprate, the minimum power level at which the OPRM should be confirmed operable is 17.6 percent of TPO RTP (i.e., 22.6 percent minus 5 percent).

The NRC staff concludes that the proposed changes with respect to the Armed Region are acceptable since the approach is consistent with the requirements specified in the DSS-CD LTR and the PBAPS plant-specific analyses.

*Backup Stability Protection (BSP)*

As discussed in Section 3.2.2 of the NRC staff's SE for the MELLLA+ amendments (Reference 7), in the event that the primary means of stability protection (DSS-CD function) becomes inoperable, there are two backup stability protection (BSP) options: (1) manual BSP and (2) automatic BSP.

The manual BSP regions and automatic BSP setpoints are confirmed or established on a cycle-specific basis. Implementation of DSS-CD consistent with the DSS-CD LTR requires that PBAPS, Units 2 and 3, confirm that the BSP approach is adequate as a part of the reload analysis.

Because PBAPS, Units 2 and 3, have implemented the DSS-CD solution consistent with the requirements of the DSS-CD LTR, the NRC staff concludes that the TPO uprate is acceptable with respect to BSP.

3.2.5 <u>TSAR Section 2.5 – Reactivity Control</u>

As discussed in TSAR Section 2.5, a plant-specific evaluation was performed for PBAPS using the evaluation approach in TLTR Section 5.6.3 and TLTR Appendix J.2.3.3. The licensee determined that there is no change in reactor pressure, temperature, or any other condition that could affect the performance of the control rod drives (CRDs) and CRD hydraulic systems and supporting equipment.

The CRD hydraulic system is independent of power level. The increased power level will have an insignificant effect on control blade depletion. The TPO uprate is not expected to change the cycle lifetime (replacement frequency) of any control blade. This factor will continue to be tracked consistent with the current PBAPS standard practice. Shutdown margin capability is included in each fuel reload evaluation.

Because the CRD system continues to meet all performance requirements at TPO uprate conditions, the NRC staff agrees that the CRD system will continue to perform all its safety-related functions at the proposed uprated conditions.

### 3.2.6 TSAR Section 2.6 – Additional Limitations and Conditions Related to Reactor Core and Fuel Performance

As discussed in TSAR Section 2.6, the licensee addressed limitations and conditions in the NRC staff's SEs for the Methods LTR (Reference 8) and the M+ LTR (Reference 9) relating to the reactor core and fuel design, as described below.

*TGBLA/PANAC Version*

The licensee used the latest versions of TGBLA and PANAC computer codes to develop the PBAPS equilibrium core for evaluation at TPO uprated conditions. The licensee confirmed that cycle-specific analyses will include the most recent TGBLA and PANAC versions as required by Methods LTR Limitation and Condition 9.1. Therefore, this limitation and condition is satisfied.

*M+ LTR SER Limitation and Condition 12.24.1*

As required by the M+ LTR Limitation and Condition 12.24.1, the licensee used the actual flow conditions to perform TRACG supporting analyses. Therefore, this limitation and condition is satisfied.

*LHGR and Exposure Qualification*

Methods LTR Limitation and Condition 9.12 states that once the PRIME computer code and its application are approved, future license applications for EPU and MELLLA+ referencing the Methods LTR (i.e., NEDC-33173P-A) must utilize the PRIME thermal-mechanical methods. The PRIME methodology was approved on January 22, 2010, and implemented in GESTAR II (Reference 45). The PBAPS MELLLA+ SAR and PBAPS TSAR are based on the GNF2 fuel product line, which has a PRIME thermal-mechanical basis. The licensee stated that PRIME fuel parameters are used in all analyses requiring fuel performance parameters, including the thermal-mechanical evaluation performed in support of the PBAPS MELLLA+ SAR and PBAPS TSAR using the PRIME thermal-mechanical methodology. Therefore, this limitation and condition is satisfied.

*GEXL-PLUS and Pressure Drop Database*

The licensee confirmed the applicability of the GNF2 experimental GEXL-PLUS and pressure drop database for operation in the MELLLA+ domain, including the extended TPO region.

The Methods LTR and the PBAPS plant-specific application of the TLTR document all analyses supporting the conclusions that the application ranges of GEH codes and methods are adequate in the MELLLA+ operating domain, including the extended TPO region. As required by M+ LTR Limitation and Condition 12.1, the range of mass fluxes and power-to-flow ratios in the GEXL-PLUS database covers the intended MELLLA+ operating domain, including the extended TPO region. The database includes low flows, high qualities, and void fractions. There are no restrictions on the application of the GEXL-PLUS correlation in the MELLLA+

operating domain, including the extended TPO region. Therefore, this limitation and condition is satisfied.

3.3 TSAR Section 3.0 – Reactor Coolant and Connected Systems

The following provides the NRC staff's technical review of the topics in Section 3.0 of the TSAR.

3.3.1 TSAR Section 3.1 – Nuclear System Pressure Relief/Overpressure Protection

The safety relief valves (SRVs) provide overpressure protection for the nuclear steam supply system (NSSS) during abnormal operational transients. As discussed in TSAR Section 5.2.1, the licensee stated that the steam flow associated with the TPO uprate can be regulated adequately by adjusting the turbine control valve (TCV) position; therefore, the operating dome pressure will not increase, and the SRV setpoints and the number of valve actuation groups will not be changed.

Evaluations and analyses for PBAPS account for 102 percent of CLTP to demonstrate that the reactor vessel conformed to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) and plant TS requirements. There is no increase in nominal operating pressure for the PBAPS TPO uprate. There are no changes in the SRV setpoints or valve out-of-service options. There is no change in the methodology or the limiting overpressure event.

The licensee stated that the analysis for each fuel reload will confirm the capability of the system to meet the ASME design criteria. Since the SRVs will actuate at the current setpoints, the current ASME overpressure protection analysis is based on operation at 102 percent power, and the ASME overpressure analysis for the first TPO uprate cycle will also account for ≥ 102 percent of CLTP, the NRC staff agrees with the licensee's assessment that the SRVs will have sufficient capacity to handle the increased steam flow associated with the proposed uprate.

3.3.2 TSAR Section 3.2 – Reactor Vessel

The NRC staff's review of the proposed MUR power uprate regarding reactor pressure vessel (RPV) integrity, addressed the following technical areas: (1) neutron fluence calculations, (2) RPV material surveillance program, (3) reactor coolant system (RCS) pressure-temperature (P-T) limits, (4) RPV beltline Charpy upper shelf energy (USE) evaluation, (5) analytical basis for permanent elimination of RPV circumferential shell weld inservice examinations pursuant to 10 CFR 50.55a(z)(1), and (6) RPV structural evaluation. Each of these technical areas is addressed in the following sections.

*Neutron Fluence Calculations*

Section 3.2.1 of the TSAR states, in part, that, "The neutron fluence for TPO was calculated by **[[** **]]**." **[[** **]]** is acceptable, as long as macroscopic cross-section data for the various water regions[1] modeled in the core does not change significantly with other

[1] Water is the only material subject to change appreciably in the context of determining a change in neutron fluence due to changing from CLTP to TPO conditions.

things being equal. More specifically, the licensee must demonstrate that material temperatures and atom densities for the various water regions in the core do not change significantly. Consequently, the only major effect on the previously calculated neutron fluence due to a change from CLTP to TPO conditions is the increase in neutron source term, which will change in direct proportion to the increase in power.[2]

In a request for additional information (RAI), the NRC staff requested that the licensee confirm that the coolant density distribution throughout the core at the proposed TPO/MELLLA+ conditions is either bounded or does not decrease significantly relative to the current licensing basis conditions defined by CLTP/MELLLA+. The licensee's response in the supplement dated August 30, 2017, stated that, "At TPO/MELLLA+ conditions the maximum [power-to-flow] ratio is reduced and is, therefore, bounded by the CLTP/MELLLA+ condition." That is, for the lowest allowable core flow at the highest allowable core power under TPO/MELLLA+ conditions, coolant density will increase slightly, which would result in an overall decrease in accumulated reactor vessel neutron fluence relative to current licensing basis values at CLTP/MELLLA+ conditions. The slight increase in coolant density is corroborated by examination of TSAR Figure 2-3, which shows that the PBAPS TSAR peak bundle exit voiding is slightly less than that of the PBAPS MELLLA+.

Coolant density perturbations due to coolant temperature changes for PBAPS as a result of the TPO are expected to be relatively small based on the proposed thermal-hydraulic parameters provided in TSAR Table 1-2, "Thermal-Hydraulic Parameters at TPO Uprate Conditions." Therefore, it is reasonable to expect that there will be no significant change in PBAPS' reactor vessel neutron fluence from CLTP to TPO conditions. The NRC staff concludes that the PBAPS fluence evaluation is acceptable based on: (1) proper accounting of the increase in neutron source term at the proposed TPO conditions, and (2) confirmation that the coolant density distribution throughout the core at the proposed TPO conditions is bounding.

*RPV Material Surveillance Program*

The RPV material surveillance program provides a means for monitoring the fracture toughness of the RPV beltline materials to support analyses for ensuring the structural integrity of ferritic components of the RPV. The NRC staff's review addressed the effects of the proposed MUR on the licensee's RPV surveillance capsule withdrawal schedule. The NRC's acceptance criteria for RPV material surveillance programs are based on: (1) final GDC 14, which requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture and of gross rupture; (2) final GDC 31, 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 non-brittle manner, and the probability of rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, which establishes requirements for monitoring changes in the fracture toughness properties of materials in the RPV beltline region; and (4) 10 CFR 50.60, which, in part, requires compliance with the material surveillance program requirements for the RCPB set forth in 10 CFR Part 50, Appendix H.

[2] See Equation 22 from R. E. Maerker, M. L. Williams, & B. L. Broadhead, *Accounting for Changing Source Distributions in Light Water Reactor Surveillance Dosimetry Analysis*, Nuclear Science and Engineering, Volume 94, Issue 4, 1986.

As an alternative to a plant-specific RPV material surveillance program, 10 CFR Part 50, Appendix H, allows for the implementation of an integrated surveillance program (ISP). An ISP is defined in 10 CFR Part 50, Appendix H, as occurring when "the representative materials chosen for surveillance for a reactor are irradiated in one or more reactors that have similar design and operating features."

The licensee discussed the impact of the proposed MUR on the PBAPS, Units 2 and 3, RPV material surveillance program in TSAR Section 3.2.1. The licensee identified that PBAPS, Units 2 and 3, participate in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) ISP. The licensee stated that prior to the ISP, the original plant-specific surveillance program consisted of three capsules at each unit. The licensee identified that the first surveillance capsule was removed from each of the PBAPS RPVs and tested after approximately 7.5 effective full power years (EFPY) prior to the implementation of the BWRVIP ISP. The licensee indicated that PBAPS, Unit 2, is designated a "host plant" for capsules under the ISP, and its representative capsules will be pulled in accordance with the ISP withdrawal schedule. The licensee indicated that PBAPS, Unit 3, is not a host plant, and both its second and third capsules are designated as standby capsules under the ISP. The licensee determined that the slight increase in the projected neutron fluence for the proposed MUR will not affect the existing ISP capsule withdrawal schedule or continued ISP implementation for either unit.

Electric Power Research Institute (EPRI) Topical Report, BWRVIP-86NP, Revision 1-A (Reference 18), establishes the ISP requirements for RPV materials (base metal and weld metal) in all operating BWRs for the first 40-year operating period and for the first 20-year period of extended operation. The NRC's final SE for BWRVIP-86NP, Revision 1-A, dated October 20, 2011 (Reference 19), documents that the BWRVIP ISP is compliant with the ISP requirements established in 10 CFR Part 50, Appendix H, for the original 40-year license terms and the first 20-year extended license terms.

The BWRVIP ISP provides for a number of surveillance capsules to be removed from specified BWRs and to be available for testing during the license renewal period for the BWR fleet. The ISP establishes acceptable technical criteria for capsule withdrawal and testing. The NRC staff identified that PBAPS, Unit 2, is a designated host plant for surveillance capsules under the BWRVIP ISP. The NRC staff verified that the small increase in neutron fluence for MUR conditions will not invalidate the estimated projected irradiation of the representative ISP capsule materials being hosted at PBAPS, Unit 2, at the specified future withdrawal times, as approved by the staff in BWRVIP-86NP, Revision 1-A. Therefore, the staff determined that the licensee's ISP capsule withdrawal schedule for PBAPS, Unit 2, will remain acceptable under MUR conditions. The staff confirmed that PBAPS, Unit 3, does not host any ISP surveillance capsules; therefore, the evaluation of a unit-specific withdrawal schedule is not applicable to Unit 3. For surveillance capsule materials being irradiated at other BWRVIP ISP host plants that were designated as "representative" for PBAPS, the NRC staff verified that the slight increase in projected neutron fluence for the PBAPS RPVs will not invalidate the corresponding irradiation of the offsite capsule materials for providing meaningful embrittlement data, as approved in BWRVIP-86NP, Revision 1-A. Therefore, the BWRVIP ISP capsule withdrawal schedule for these offsite capsules will remain acceptable, considering the slight increase in projected irradiation of the PBAPS RPVs under the proposed MUR conditions. The NRC staff identified that ISP capsule materials will continue to be appropriately handled and tested in accordance with the BWRVIP ISP criteria, as approved by the NRC, for meeting the requirements of 10 CFR Part 50, Appendix H. Therefore, the NRC staff determined that the licensee's RPV material surveillance program for PBAPS, Units 2 and 3, is acceptable for satisfying the requirements of 10 CFR Part 50, Appendix H, for MUR conditions.

*Pressure-Temperature Limits and Upper Shelf Energy Requirements and Guidance*

Appendix G to 10 CFR Part 50 provides fracture toughness requirements for ferritic materials in the RCPB, including requirements for the Charpy USE for protecting RPV beltline materials against non-brittle failure and requirements for calculating RCS P-T limits for protection against brittle fracture. The RCS P-T limits are specifically established to ensure the structural integrity of the ferritic components of the RCPB (in particular the RPV) during any condition of normal operation, including AOOs and hydrostatic tests. The NRC staff's review of USE and P-T limits addressed the licensee's current licensing basis methodologies for USE (or mandated alternative analyses discussed below) and P-T limits, and its plant-specific evaluation for demonstrating that 10 CFR Part 50, Appendix G requirements will continue to be satisfied following implementation of the proposed MUR, specifically considering neutron embrittlement of RPV beltline materials for MUR conditions.

The NRC's acceptance criteria for USE and P-T limits are based on: (1) final GDC 14, which requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture and of gross rupture; (2) final GDC 31, 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 non-brittle manner, and the probability of rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which provides fracture toughness requirements for ferritic materials in the RCPB; and (4) 10 CFR 50.60, which, in part, requires compliance with the fracture toughness requirements for the RCPB set forth in 10 CFR Part 50, Appendix G.

Section IV.A.1 of 10 CFR Part 50, Appendix G, provides requirements for maintaining acceptable levels of USE for RPV beltline materials throughout the licensed operating terms of nuclear power reactors. The rule requires that RPV beltline materials have Charpy USE in the transverse direction for base material and along the weld for weld material greater than or equal to 75 foot-pounds (ft-lbs) in the unirradiated condition. The rule also requires that RPV beltline materials must maintain Charpy USE greater than or equal to 50 ft-lbs throughout the operating life of the RPV, unless it is demonstrated in a manner approved by the NRC that lower values of USE would provide margins of safety against fracture equivalent to those required by the ASME Code, Section XI, Appendix G. The analysis to demonstrate acceptable margins of safety against fracture is generally referred to as an "Equivalent Margins Analysis" (EMA). The rule also requires that the methods used to calculate projected USE values or perform EMAs must account for the effects of neutron radiation on the USE values or EMA results for the materials and must incorporate any credible RPV surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H, RPV material surveillance program. The NRC staff's recommended guidelines for calculating the effects of neutron radiation on the USE values for the RPV beltline materials are provided in NRC Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, dated May 1988 (Reference 20).

Section IV.A.2 of 10 CFR Part 50, Appendix G, requires that the P-T limits for operating reactors be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of the ASME Code, Section XI, Appendix G. The rule also requires that the

P-T limits calculations account for the effects of neutron radiation on the material properties of the RPV beltline materials and that P-T limits calculations incorporate any applicable RPV surveillance capsule data that are reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, RPV materials surveillance program. The ASME Code, Section XI, Appendix G, specifies a procedure for calculating P-T limits that is based on the principles of linear elastic fracture mechanics. The key material property input to this linear elastic fracture mechanics procedure is the critical stress intensity factor, $K_{IC}$, also referred to as the fracture toughness. $K_{IC}$ is a function of the difference in metal temperature and the reference nil-ductility temperature ($RT_{NDT}$) for the material. Therefore, for a given $RT_{NDT}$ value, $K_{IC}$ is a single-valued function of metal temperature, referred to as the $K_{IC}$ curve. Neutron irradiation of RPV beltline materials will increase their $RT_{NDT}$ values, thereby causing a "shift" to the $K_{IC}$ vs. temperature curve, which directly corresponds to a conservative shift in the RPV beltline P-T limit curve. The NRC staff's recommended guidelines for calculating the effects of neutron radiation on the $RT_{NDT}$ for RPV beltline materials, whereby licensees calculate the "adjusted $RT_{NDT}$" (ART) value due to neutron radiation, are specified in RG 1.99, Revision 2. Finally, recent guidelines published in RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (Reference 21), provide additional NRC staff expectations for evaluations of P-T limits in licensing applications, including guidance on the consideration structural discontinuities[3] in the development of P-T limits. As discussed in the RIS, structural discontinuities could potentially result in more bounding P-T limits than those defined by the RPV beltline shell material with the most limiting fracture toughness. Therefore, structural discontinuities need to be considered in the development of P-T limit curves.

*USE Evaluation*

In TSAR Section 3.2.1, the licensee stated that the USE will remain greater than 50 ft-lb for the design life of the RPV or its EMA results will satisfy the requisite margins against fracture required by 10 CFR Part 50, Appendix G. The licensee identified that many of the PBAPS, Units 2 and 3, RPV materials do not have sufficient unirradiated USE data and, therefore, require an EMA. The licensee stated that the EMA was performed for the limiting beltline plate and weld materials to ensure compliance. The licensee provided the USE values and EMA calculations in Tables 3-1a and 3-1b of the TSAR. These tables show calculations of projected USE and EMA results through 54 EFPY for MUR conditions, based on RG 1.99, Revision 2.

RG 1.99, Revision 2. recommends that projected USE values for RPV beltline materials be calculated based on the projected neutron fluence at a postulated flaw depth corresponding to one-quarter of the RPV beltline wall thickness from the clad/base metal interface of the RPV (1/4T location), the weight percentage (wt. %) of copper (Cu) in the material, and the preservice (unirradiated) USE value for the material. The 1/4T projected neutron fluence and the wt. % Cu are used to calculate the projected percentage decrease in the USE for the material. If valid unirradiated USE data is available, the projected percentage decrease in USE directly determines the projected USE at the end of the licensed operating term. If projected USE

[3] "Structural discontinuities" are defined as those RPV components that have significantly elevated states of stress relative to the RPV beltline shell region. This includes complex geometry RPV components such as nozzles, flanges, and RPV shell regions near these (e.g., nozzle welds), wherein the tensile stress that would propagate a postulated flaw is significantly greater than that for the limiting RPV beltline shell material (i.e., the beltline material with the highest adjusted reference temperature (ART)). Such RPV components are also referred to as "geometric discontinuities" in the ASME Code, Section XI, Appendix G, G-2220.

values do not meet the 50 ft-lbs acceptance criterion from 10 CFR Part 50, Appendix G, or if plant-specific unirradiated USE data for the material is not available or is not adequate to demonstrate projected USE greater than 50 ft-lbs, an EMA shall be performed for the licensed operating term.

Like most BWRs, the original certified RPV material fabrication records for PBAPS, Units 2 and 3, do not include adequate Charpy USE data for the unirradiated condition. These BWR plants are, therefore, unable to demonstrate that the USE for their RPV beltline materials will be maintained greater than 50 ft-lbs through the licensed operating terms. Accordingly, these BWR plants have been required to perform EMAs for 40-year license terms and for 20-year extended license terms, in order to demonstrate that the RPV beltline materials will remain in compliance with 10 CFR Part 50, Appendix G requirements. Topical Report NEDO-32205-A, Revision 1 (Reference 22), documents the original (40-year) generic EMA methodology and EMA results that were approved by the NRC-staff for generic application to BWRs for demonstrating compliance with 10 CFR Part 50, Appendix G. These generic EMA results for BWRs were later updated for license renewal (LR) via the BWRVIP-74-A report (Reference 23), which was reviewed and approved by the NRC staff on October 18, 2001 (Reference 24) for referencing in LR applications. The BWRVIP-74-A report identifies that LR applicants must demonstrate that their RPV beltline materials (i.e., RPV materials with projected neutron fluence exposure greater than $1 \times 10^{17}$ n/cm$^2$ (E > 1.0 MeV), as addressed in RIS 2014-11) satisfy the applicable generic EMA acceptance criteria from the NEDO report on a plant-specific basis in applications for LR. LR applicants addressed the NRC staff-approved criteria in BWRVIP-74-A by (1) performing plant-specific RG 1.99, Revision 2 calculations of the projected percentage decrease in the USE for the limiting RPV beltline plate and weld materials for 20-year extended license terms; and (2) then comparing the projected percentage decrease in USE to the applicable EMA acceptance criterion for the BWR plant categories and plate and weld material types, as specified in BWRVIP-74-A.

Accordingly, for all BWR power uprate applications, licensees must reevaluate the projected percentage decrease in the USE for the limiting plate and weld materials based on the wt. % Cu content for the material and the projected neutron fluence at the 1/4T location for the proposed power uprate conditions. The licensee provided the limiting RPV beltline plate and weld EMA results for MUR conditions in Tables 3-1a and 3-1b of the TSAR for PBAPS, Units 2 and 3, respectively. The NRC staff independently verified that the licensee's calculations of projected percentage decrease in USE for the limiting RPV beltline plate and weld were correctly performed in accordance with RG 1.99, Revision 2. The NRC staff confirmed that wt. % Cu values used for these calculations are consistent with current licensing basis values that were previously accepted by the staff in its SE for previous PBAPS, Units 2 and 3, license amendments to implement the P-T Limits Report (PTLR) in April 2013 (Reference 25) and the EPU in August 2014 (Reference 4). The staff also determined that the neutron fluence inputs to the licensee's calculation of projected percentage decrease in USE are acceptable for the proposed MUR conditions based on the fact that licensee used NRC-approved neutron fluence calculational methods that are consistent with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 26). Finally, the staff also verified that the licensee used the correct EMA acceptance criteria from BWRVIP-74-A for demonstrating that the projected percentage decrease in USE for its limiting RPV beltline plate and weld materials will remain acceptable for the period of extended operation under MUR conditions. Therefore, the staff determined that the licensee's EMA results for the limiting RPV beltline plate and weld materials at PBAPS, Units 2 and 3, are acceptable for satisfying the requirements of 10 CFR Part 50, Appendix G, for MUR conditions.

In addition to limiting RPV beltline plate and weld EMA results for MUR conditions, the licensee also provided the ISP surveillance plate and weld data for the PBAPS, Units 2 and 3, RPV beltline material USE evaluation in Tables 3-1a and 3-1b of the TSAR. The staff noted that the ISP materials for PBAPS, Units 2 and 3, are not the most limiting RPV beltline materials relative to the projected percentage decrease in USE. The EMA results for the limiting materials discussed above are the primary concern for the USE evaluation. However, the staff reviewed the licensee's application of this ISP data to the USE evaluations and verified that the data were correctly applied to validate the USE results for these materials, consistent with RG 1.99, Revision 2, and in accordance with 10 CFR Part 50, Appendix G, for MUR conditions.

*P-T Limits Evaluation*

In Section 3.2.1 of the TSAR, the licensee identified that the projected ART values for the RPV beltline materials have been evaluated for the proposed MUR. A listing of the 54 EFPY ART values and all input parameters, which include material property inputs and projected neutron fluence, are provided in Tables 3-2a and 3-2b of the TSAR. The licensee stated that the proposed MUR conditions will result in a slight increase in neutron fluence for 54 EFPY; therefore, the licensee's projected 54 EFPY ART values for MUR conditions are generally projected to increase by a small amount, on the order of 0.5 degrees Fahrenheit (°F).

The licensee indicated that considering the slight increase in the projected 54 EFPY ART values, the current RPV beltline P-T limit curves will remain valid for MUR conditions, provided that an adjustment to the EFPY applicability term for the P-T limits is implemented. The licensee identified that to ensure that the P-T limits remain bounding for MUR conditions, the EFPY applicability term for the P-T limits will be conservatively adjusted downward from 54 EFPY to 53 EFPY. The licensee stated that based on plant operating time thus far, the total projected operating time for the 60-year renewed license will remain less than 53 EFPY.

The NRC staff's review of the licensee's P-T limits addressed the continued validity of the licensee's current licensing basis P-T limits methodology for MUR conditions and the effects of the proposed MUR on neutron embrittlement of RPV beltline materials.

By letter dated April 1, 2013 (Reference 25), the NRC staff issued Amendment Nos. 286 and 289 for PBAPS, Units 2 and 3, respectively, authorizing the relocation of the PBAPS, Units 2 and 3, P-T limits from the TSs to a new licensee-controlled document called the P-T Limits Report (PTLR), consistent with the guidance in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protections System Limits" (Reference 27). These amendments also added new administrative controls via TS 5.6.7, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)." TS 5.6.7 provides requirements for the control of changes to the plant-specific P-T limits and for submittal of PTLR revisions to the NRC. TS 5.6.7 states that the analytical methods used to determine the P-T limits shall be those previously reviewed by the NRC as described in GEH Topical Report NEDO-33178-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 1, dated June 2009 (Reference 28). The NRC staff's SE accompanying these license amendments documents the bases for acceptance of the licensee's implementation of the GEH PTLR methodology for generating current and future plant-specific P-T limits in accordance with the TS 5.6.7 administrative controls.

The current licensing basis 54 EFPY P-T limits for PBAPS, Units 2 and 3, were calculated based on the licensee's plant-specific application of the methodology documented in Topical Report NEDO-33178-A. The topical report includes the NRC staff's SE authorizing its use in developing plant-specific PTLRs for BWRs. As noted above, the GEH methodology is directly referenced in the TS 5.6.7 administrative controls for each unit, and it thus constitutes the TS requirements for controlling future changes to the P-T limit curves and associated parameters contained in the PTLR. Any change to the analytical methods to determine the P-T limits that deviates from the GEH methodology cited in TS 5.6.7 would require prior NRC approval by license amendment pursuant to 10 CFR 50.90.

As documented in its SE for PBAPB, Units 2 and 3, Amendment Nos. 286 and 289, the NRC staff had previously verified that the licensee's P-T limit calculations based on the GEH PTLR methodology adequately addressed and were bounding for all ferritic RPV beltline and non-beltline RPV components, including those structural discontinuities that could potentially result in more bounding P-T limits than those generated for the limiting RPV beltline shell material. The NRC staff specifically notes that the consideration of beltline and non-beltline RPV structural discontinuities for generating P-T limits is necessary for demonstration of compliance with 10 CFR Part 50, Appendix G, as addressed in RIS 2014-11. Based on these considerations, the NRC staff determined that the current TS-controlled generic methodology for generating the PBAPS, Units 2 and 3, P-T limits will remain acceptable for MUR conditions.

For its evaluation of the specific effects of the MUR on the PBAPS, Units 2 and 3, P-T limits for the RPV beltline region, the NRC staff reviewed and independently verified the licensee's RPV beltline material ART calculations for MUR conditions, as provided in Tables 3-2a and 3-2b of the TSAR. The NRC staff confirmed that the 54 ART values for the RPV beltline plates, welds, and nozzles were correctly recalculated for MUR conditions in accordance with the recommended guidelines of RG 1.99, Revision 2, taking into consideration the slight increase in the projected neutron fluence that would result from the MUR. The NRC staff verified that all RPV beltline material property inputs to the ART calculations, as listed in Tables 3-2a and 3-2b of the TSAR, are consistent with current licensing basis values that were previously accepted by the staff as part of its evaluation for the amendments to implement the PBAPS PTLR (Reference 25) and the EPU (Reference 4). The staff also determined that the neutron fluence inputs to the ART calculations are acceptable for the proposed MUR conditions based on the fact that the licensee used NRC-approved neutron fluence calculational methods that are consistent with RG 1.190.

The NRC staff finds that TS 5.6.7 provides the necessary administrative controls for updating the PTLR to account for new neutron fluence-based operating periods and for submitting these PTLR revisions to the NRC. This includes, among other things, any changes to neutron fluence periods that are calculated based on implementation of power uprates. The NRC staff determined that the actual P-T limit curves and the specific RPV beltline neutron fluence inputs to the ART calculation for the P-T limits do not require any revision, provided that appropriate adjustments to the EFPY applicability period corresponding to existing neutron fluence values and ARTs are implemented in the PTLR to account for MUR conditions. As discussed above, the licensee determined the proposed MUR would result in a slight increase in the projected 54 EFPY ART values that is on the order of 0.5 °F. The licensee also determined that a decrease in the EFPY applicability term from 54 EFPY to 53 EFPY for the existing P-T limits, ARTs, and neutron fluence inputs contained in the PTLR would ensure that the current licensing basis P-T limits remain bounding for MUR conditions. The NRC staff verified that such an adjustment is conservative for all RPV beltline materials, and the adjustment will ensure that the existing PTLR P-T limits and ART values will remain in compliance with 10 CFR Part 50, Appendix G requirements for MUR conditions. The staff finds that TS 5.6.7 provides the necessary controls to ensure that this change to the EFPY period for the neutron fluence, ARTs, and P-T limits will be implemented in the PTLR following the completion of the NRC staff's review of the MUR application and implementation of the MUR at the plant. Therefore, the staff determined that the licensee's evaluation of its P-T limits and RPV beltline region ART values for MUR conditions is acceptable for satisfying the requirements of 10 CFR Part 50, Appendix G.

*RPV Circumferential and Axial Welds*

The inservice inspection (ISI) of ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable Addenda, as required by 10 CFR 50.55a(g). The ASME Code, Section XI, Table IWB-2500-1, requires inspection of all RPV welds every 10-year ISI interval. The NRC may grant relief from specific ISI requirements in accordance with 10 CFR 50.55a(g)(6)(i) based on licensees' demonstrated impracticalities experienced at the plant when attempting to perform the ASME Code Section XI examinations. In addition, the NRC can authorize plant-specific alternatives to these requirements in accordance with 10 CFR 50.55a(z). Pursuant to 10 CFR 50.55a(z), plant-specific alternatives to the requirements of paragraph (g) may be used when authorized by the NRC if (1) the proposed alternatives would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

In Section 3.2.1 of the TSAR, the licensee stated, in part, that the limiting 54 EFPY beltline circumferential and axial weld mean $RT_{NDT}$ values will remain bounded by the criteria in NRC-approved Topical Reports BWRVIP-05 (Reference 29) and BWRVIP-74-A (Reference 23). The licensee provided the results of the limiting RPV beltline circumferential weld evaluation in Tables 3-3a and 3-3b of the TSAR for PBAPS, Units 2 and 3, respectively. The licensee provided the results of the limiting RPV beltline axial weld evaluation in Tables 3-4a and 3-4b of the TSAR for PBAPS, Units 2 and 3, respectively.

Alternatives allowing for permanent elimination of RPV circumferential shell weld exams at BWRs were authorized by the NRC for specific plants pursuant to 10 CFR 50.55a(z)(1) for the duration of the plants' original 40-year licensed operating terms, based on the licensees' application of the NRC-approved BWRVIP probabilistic fracture mechanics (PFM) methodologies that are described in the staff's SE for BWRVIP-05. For 20-year extended license terms, BWR licensees must re-apply for these ASME Code alternatives to allow for the application of the staff-approved BWRVIP PFM methods to justify elimination of RPV circumferential shell weld exams for periods of extended operation. The NRC-approved BWRVIP-74-A report provides the technical basis for application of the PFM methods for periods of extended operation. The NRC staff's SE accompanying the BWRVIP-74-A report provides the NRC staff's specific RPV weld neutron embrittlement acceptance criteria for plant-specific application of these PFM results for elimination of BWR RPV circumferential shell examinations for 20-year extended license terms. These acceptance criteria must be satisfied for both RPV circumferential and axial welds[4]. Accordingly, licensees shall demonstrate in their 10 CFR 50.55a(z)(1) submittals that the projected embrittlement of their limiting RPV circumferential and axial welds are bounded by the acceptance criteria documented in the BWRVIP-74-A SE for periods of extended operation. As documented in the BWRVIP-74-A SE, the projected embrittlement is determined based on the mean $RT_{NDT}$ values for the limiting circumferential and axial welds, which is equal to the unirradiated $RT_{NDT}$ value, plus the projected mean value of the shift in $RT_{NDT}$ for the period of extended operation. The mean value of the shift in the $RT_{NDT}$ is calculated at the interface of the RPV clad and weld metal in accordance with the procedures of RG 1.99, Revision 2.

By letter dated January 24, 2012 (Reference 31), the NRC authorized the licensee's use of the above BWRVIP PFM methodology as a plant-specific alternative to the ASME Code, Section XI, RPV circumferential shell weld examinations at PBAPS, Units 2 and 3, for the period of extended operation. Tables 3-3a, 3-3b, 3-4a, and 3-4b of the TSAR provide the necessary update for the limiting RPV circumferential and axial weld mean $RT_{NDT}$ values that are the plant-specific basis for this alternative for MUR conditions. The NRC staff reviewed the licensee's mean $RT_{NDT}$ calculations for MUR conditions and determined that they are correct and represent the most limiting RPV circumferential and axial weld materials for PBAPS, Units 2 and 3. The staff reviewed the initial $RT_{NDT}$ values, copper content percentages, and nickel content percentages that were used as inputs for the mean $RT_{NDT}$ calculations and verified that they are consistent with those previously approved by the staff for the PTLR and EPU amendments. The staff also determined that the neutron fluence inputs to the mean $RT_{NDT}$ calculations are acceptable for the proposed MUR conditions based on the licensee's use of NRC-approved neutron fluence calculational methods that are consistent with RG 1.190. Finally the staff verified that the licensee applied the correct NRC acceptance criteria for the circumferential and axial weld mean $RT_{NDT}$ values from the BWRVIP-74-A SE for determining that the welds will continue to satisfy the applicable RPV PFM results supporting the elimination of the RPV circumferential shell weld exams for the period of extended operation under MUR conditions. Therefore, the staff determined that the licensee's evaluation of its limiting RPV circumferential and axial welds for MUR conditions is acceptable for ensuring that its analytical basis for permanent elimination of RPV circumferential shell weld exams remains valid for the period of extended operation. The staff notes that the NRC-authorized alternative (Reference 31) is only applicable for elimination of RPV circumferential shell weld exams. Volumetric examination of RPV axial welds, including locations where they intersect with RPV circumferential welds, is still required every 10-year ISI interval during the period of extended operation at PBAPS, Units 2 and 3, in accordance with the requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g).

*RPV Structural Evaluation*

In TSAR Section 3.2.2, the licensee discussed the results of its structural evaluation of the RPV for the TPO uprate. This included a bounding reconciliation for stresses for normal, upset,

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[4] The NRC staff's 1998 SE for BWRVIP-05 identified a need to further evaluate the higher conditional failure probability levels for RPV axial welds. Therefore, the staff performed a review of supplemental BWRVIP correspondence regarding the axial weld failure probabilities. The staff's March 7, 2000, supplemental SE for BWRVIP-05 (Reference 30) concluded that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet are acceptable on a generic basis; however, these generic axial weld results are only applicable to 40-year license terms. Therefore, consideration of BWR axial welds for renewed license terms would require a plant-specific treatment, based on plant-specific evaluation of limiting axial weld embrittlement for periods of extended operation.

emergency, and faulted conditions, as well as fatigue for normal and upset conditions. To address concerns pertaining to several GEH safety communication letters, the licensee demonstrated, as shown in TSAR Table 3-5, that the shroud support attachment to the RPV component is within the allowable stress and fatigue limits, including acoustic loads, and is, therefore, structurally qualified for operation at TPO uprate conditions.

The licensee's reconciliation is a bounding reconciliation for stress and fatigue of the reactor vessel, considering a 60-year plant license, because the evaluation **[[**

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Based on a review of the licensee's evaluation, the NRC staff concludes that the reactor vessel will maintain its structural integrity allowable limits per the applicable ASME Code, 1965 Edition, including Winter 1965 Addenda, from stress and fatigue considerations for normal and upset conditions, as well as from stress considerations for emergency and faulted conditions, at the proposed TPO uprate conditions.

*Reactor Vessel Conclusion*

Based on the considerations discussed above, the NRC staff concludes that there is reasonable assurance that the structural integrity of the RPV will continue to be maintained under TPO uprate conditions, consistent with the regulatory requirements set forth in 10 CFR Part 50, Appendix G; 10 CFR Part 50, Appendix H; 10 CFR 50.60; and 10 CFR 50.55a.

### 3.3.3 TSAR Section 3.3 – Reactor Internals

As discussed in TSAR Section 3.3, the reactor internals include: (1) core support structure (CSS) components (shroud, shroud support, core plate, top guide, control rod drive housing, control rod guide tube, orificed fuel support, and fuel channel); and (2) non-CSS components (feedwater sparger, jet pump assembly, core spray line and sparger, access hole cover, steam dryer, shroud head and steam separator assembly, in-core housing and guide tube, core differential pressure and liquid control line, and jet pump instrumentation penetration seal). The licensee's evaluation included the following areas for the reactor internals: (1) reactor internal pressure differences; (2) reactor internals structural evaluation; and (3) steam separator and dryer performance. Each of these areas is discussed below.

*Reactor Internal Pressure Differences and Reactor Internals Structural Evaluation*

The reactor internals are not ASME Code components, but the licensee utilized the ASME Code as guidance in the structural evaluations of the internals. The reactor internal pressure differences (RIPDs) for the TPO uprate for normal and upset conditions are slightly higher than CLTP (which includes EPU and MELLLA+). However, the reconciliation evaluations demonstrated that all reactor internals are within the allowable limits. The limiting stresses and fatigue usage factors for all RPV internals are shown to be acceptable. The emergency and faulted evaluations of RIPDs for the TPO uprate are conservatively bounded by the EPU analysis performed at 102 percent of CLTP.

The appropriate loads are considered by the licensee with applicable load combinations in the evaluation of the RPV internals, which include RIPDs, dead weight, seismic loads, acoustic and flow-induced loads due to recirculation line break (RLB), hydraulic flow, and thermal loads. The RPV design pressure remains unchanged. RIPD loads are bounded by the analysis performed at 102 percent of CLTP except for the shroud. The slight increase in the shroud RIPD is insignificant. The seismic load and acoustic and flow-induced loads due to RLB are also unchanged for TPO. The feedwater (FW) flow and thermal load remain bounded by the 102 percent of CLTP values. The change in hydraulic flow is considered negligible. The effect of weight change on load due to jet pump repair is also insignificant.

Based on a review of the licensee's evaluation, the NRC staff concludes that the reactor internals will maintain their structural integrity from stress and fatigue considerations for normal and upset conditions, as well as from stress considerations for emergency and faulted conditions, at the proposed MUR power level.

*Steam Separator and Dryer Performance*

As discussed in TSAR Section 3.3.3, the licensee performed a plant-specific evaluation for PBAPS steam separator and steam dryer performance. TPO will result in an increase in the amount of saturated steam generated in the reactor core. For constant core flow, this increase in the amount of saturated steam results in an increase in the average separator inlet quality and an increase in the steam dryer face velocity. The results of the evaluation demonstrated that the steam separator/dryer performance will be acceptable because the moisture content will remain less than or equal to the current design basis value of 0.10 weight percent at TPO conditions.

Attachment 10 to the application dated February 17, 2017, provided Westinghouse proprietary report LTR-BWR-ENG-16-032-P, "Peach Bottom Units 2 and 3, Steam Dryer Report at MUR Conditions," Revision 0. A non-proprietary (i.e., publicly available) version of the report was included as Attachment 12 to the application.

As discussed in the Westinghouse report, for a high-cycle fatigue assessment of the steam dryers, a complete reanalysis was performed using the Westinghouse steam dryer acoustic/structural methodology. The assessment used main steam line data extrapolated to MUR conditions, which was analyzed using the Acoustic Circuit Enhanced (ACE) Revision 3.1 software computer program. ACE Revision 3.1 includes the end-to-end biases and uncertainties from the PBAPS, Unit 2, benchmarking performed at EPU conditions. Additionally, the reanalysis considered the effects of non-main steam line acoustic loads. The effects of MELLLA+ conditions have also been assessed. The resulting minimum alternating stress ratios (MASRs), as shown in the table below for the steam dryer limiting components, are greater than 1.0, demonstrating that the high-cycle fatigue stresses are within acceptable limits based on the ASME Code that was used for guidance. The MASR is defined as 13,600 pounds per square inch (psi) divided by the maximum computed alternating stress. The 13,600 psi value corresponds to the ASME Code high-cycle fatigue stress limit corresponding to $10^{11}$ cycles for stainless steel.

| PBAPS Unit | Dryer Location | Component | MASR for Combined EPU, MELLLA+, and MUR |
|---|---|---|---|
| Unit 2 | [[ ]] | [[ ]] | [[ ]] |
| Unit 2 | [[ ]] | [[ ]] | [[ ]] |
| Unit 3 | [[ ]] | [[ ]] | [[ ]] |
| Unit 3 | [[ ]] | [[ ]] | [[ ]] |

In addition to the high-cycle fatigue assessment, the PBAPS steam dryers were also evaluated by the licensee to show compliance with the structural requirements of the ASME Code, Section III, Subsection NG. The assessment shows that the dryers meet the stress and fatigue usage limits of the ASME Code for the duty cycles due to implementation of MUR for normal operation (Service Level A), upset conditions (Service Level B), emergency conditions (Service Level C), and faulted conditions (Service Level D).

The NRC staff concludes that since the licensee's evaluation indicates that the steam dryers will meet the applicable allowable limits, there is reasonable assurance the steam dryers will maintain their structural integrity under the proposed MUR conditions.

3.3.4 <u>TSAR Section 3.4 – Flow-Induced Vibration</u>

As discussed in TSAR Section 3.4, the licensee evaluated the effects of flow-induced vibration (FIV) for the reactor vessel internals. TSAR Section 3.4 also included evaluations of the effects of FIV for safety-related piping components and thermowells in the main steam (MS), feedwater (FW), and reactor recirculation system (RRS) piping.

The evaluation by the licensee determined the effects of FIV on the reactor internals at 110 percent of rated core flow (CF) and at a power level of 4,030 MWt (i.e., 102 percent of CLTP). The vibration levels for the TPO conditions were estimated from measured vibration data during startup tests on the prototype plant, Browns Ferry Nuclear Plant, Unit 1. For components requiring an evaluation but that were not instrumented at Browns Ferry Nuclear Plant, Unit 1, vibration data from similar plants or test facilities, or analytical results, were utilized. The licensee estimated expected vibration levels for TPO by extrapolating the measured vibration data based on GEH BWR operating experience. These expected vibration levels were compared with vibration acceptance limits.

For the proposed rated TPO thermal power level, there is an increase in FW flow of approximately 2 percent. This change results in less than a 4 percent increase in FIV stresses in the feedwater sparger, shroud, shroud head, and separator. By extrapolation of measured data, the licensee determined that the resulting stresses are within acceptable limits.

For the jet pumps, the increase in jet pump flow at TPO is negligible, and the evaluation determined the FIV stress would increase by less than 2 percent. By extrapolation of measured data, the licensee determined that the resulting stresses are within acceptable limits.

Since there is no change in CF from CLTP and TPO, there is no change in stresses for the control rod guide tube and the in-core guide tube. There is also no change in stress for the RPV top head nozzles because there is negligible steam flow in that area.

For the core spray piping and sparger, based on ASME Appendix N criteria, the licensee determined that there is adequate separation between the structural natural frequencies and the vortex shedding frequency. Therefore, no resonance or change in stress occurs in the TPO region.

For jet pump sensing lines, the licensee also determined that no resonance occurs at the vane passing frequency range of the recirculation pumps due to TPO.

The fuel assemblies used at PBAPS (i.e., GNF2) are acceptable for TPO conditions because the operating conditions used in the fuel assembly design are bounding for PBAPS at TPO conditions.

By conservatively assuming a lock-in condition, the maximum stresses in the guide rods were calculated and shown to be within the GEH acceptance limit of 10,000 psi. It is noted that the GEH acceptance limit of 10,000 psi is more conservative compared with the ASME Code high-cycle fatigue stress limit of 13,600 psi corresponding to $10^{11}$ cycles for stainless steel.

The flow rates in safety-related MS and FW piping increase by less than 2 percent due to the TPO uprate. The RRS flow rate is essentially unchanged at TPO. The MS and FW piping thermowells are expected to experience increased vibration levels approximately proportional to the increase in the square of the flow velocities and in proportion to any change in fluid density. As the result of a roughly 2 °F increase in FW temperature, there is an insignificant decrease in fluid density for TPO conditions. The licensee stated that analytical evaluations have shown that the safety-related piping components and thermowells in the MS, FW, and RRS piping are structurally adequate for TPO conditions.

Based on a review of the licensee's evaluations as summarized above, the NRC staff concludes that there is reasonable assurance that the structural integrity of the reactor internals and MS, FW, and RRS thermowells and piping components will not be impacted as a result of FIV under TPO conditions.

### 3.3.5 <u>TSAR Section 3.5 – Piping Evaluation</u>

*Structural Integrity and Pressure Boundary Integrity*

The licensee's evaluations for RCPB piping and its supports are included in TSAR Section 3.5.1. The licensee's evaluations for balance-of-plant (BOP) piping and its supports are included in TSAR Section 3.5.2.

The licensee described that there is no change in methods used for plant-specific piping and pipe support evaluations from those used in PBAPS EPU evaluations. The effect of the TPO uprate with no nominal vessel dome pressure increase is negligible for the RCPB portion of all piping except for portions of the FW lines, MS lines, and piping connected to the FW and MS lines. The steam flow in the MS lines at TPO RTP is approximately 2 percent higher than at CLTP. There is an increase of roughly 2 percent in FW flow and pressure and a 2 °F increase in FW temperature.

The current licensing basis analyses envelop TPO conditions for the recirculation piping due to the small change in core pressure drop (less than 1 psi) and small change in the recirculation fluid temperature (less than 1 °F). As such, there are negligible changes in piping stresses and negligible effect on pipe supports. Similarly, for the RPV bottom head line, reactor core isolation cooling piping, high-pressure coolant injection piping, core spray piping, SLC piping, and reactor water cleanup piping, the current licensing basis analyses envelop TPO conditions due to the small change in process pressure (less than 1 percent) and small change in recirculation fluid temperature (less than 1 °F). As such, there are negligible changes in piping stresses and negligible effect on pipe supports.

For the MS and FW lines, supports, and connected lines, factors were applied by the licensee to determine the percentage increases in applicable ASME code stresses, displacements, cumulative usage factors (CUFs), and pipe interface component loads (including supports, anchors, and equipment nozzle loads) as a function of the percentage increase in pressure (where applicable), temperature, and flow due to TPO conditions. The licensee applied the percentage increases to the highest calculated stresses, displacements, and CUFs at applicable piping system node points to conservatively determine the maximum TPO calculated stresses, displacements, and usage factors. This approach is conservative because the TPO does not affect certain loads such as weight and all building filtered loads (i.e., seismic loads are not affected by the TPO). The licensee determined that the MS piping, FW piping, and attached piping and pipe supports meet applicable acceptance criteria (e.g., ASME Code) and that the current licensing basis evaluation conclusions remain valid for TPO conditions.

The licensee stated that operating pressures and temperatures under TPO conditions for BOP piping outside containment (i.e., condensate, FW, extraction steam (ES), heater drain, and MS systems) will remain within design ratings. As there is no change in the MS operating temperature from the reactor to the MS stop valves, the thermal expansion stresses for TPO remain unchanged.

For systems with increased operating temperatures (i.e., MS piping downstream of the stop valves, condensate, FW, ES, and heater drains), changes to thermal expansion stresses are small and were determined to be acceptable. Pipe support loads will experience a small increase in thermal loads (less than 1 percent). However, when considering the combination with other loads that are not affected by the TPO uprate (e.g., deadweight), the combined support load increase is insignificant. For the MS system piping outside containment, the existing turbine stop valve closure transient analysis bounds the TPO uprate conditions. The BOP piping and supports (outside of the RCPB) for operation at the TPO conditions are adequate because the changes due to TPO are insignificant.

The NRC staff finds that the licensee has adequately addressed the effect of minor changes in MS flow, FW flow, FW pressure, and FW temperature due to TPO RTP on piping stresses and pipe supports. For other RCPB, as well as BOP piping and supports, there is negligible impact and the current licensing basis envelops TPO conditions. Therefore, the staff concludes that there is reasonable assurance that the RCPB piping and supports, as well as BOP piping, will continue to maintain their structural integrity and pressure boundary integrity at the TPO uprate conditions.

*Erosion/Corrosion*

In addition to the piping evaluations described above, TSAR Section 3.5 provided an evaluation of the piping systems with respect to flow-accelerated corrosion (FAC). FAC is a corrosion mechanism that occurs in carbon steel components exposed to either single-phase or two-phase water flow. Components made from stainless steel are not affected by FAC, and FAC is significantly reduced in components containing a small amount of chromium or molybdenum. The rates of material loss due to FAC depend on the system flow velocity, component geometry, fluid temperature, steam quality, oxygen content, and pH. During plant operation, it is not normally possible to maintain all of these parameters in a regime that minimizes FAC; therefore, loss of material by FAC can occur. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

In TSAR Section 3.5.1, the licensee stated that PBAPS has a program established to monitor FAC for pipe wall thinning in both single and two-phase high energy carbon steel piping in the MS, FW, and BOP systems. The licensee stated that the CHECWORKS™ SFA 3.0 predictive modeling program is used to calculate potential wall thinning of components susceptible to FAC. Additionally, the licensee stated that its FAC program has been previously evaluated by the NRC staff during the license renewal process and for the EPU at PBAPS. The program is based on compliance with GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning" (Reference 43).

In the LAR, the licensee stated that operation at the proposed MUR power uprate conditions results in changes to parameters (e.g., flow velocity, temperature, and moisture content) in certain systems. The licensee stated its evaluation of predicted wall thinning indicates a minimal effect for the MS, FW, and attached systems, but that the FAC monitoring program will continue to consider adjustments related to predicted material loss. The licensee stated that high energy piping systems will continue to be monitored in order to provide confidence in the integrity of these systems. In addition, the licensee stated that no changes to inspection frequencies are required to maintain adequate margin for certain piping systems. In addition, the licensee stated that the continued use of existing procedures will provide reasonable assurance that any adverse effect from the TPO on FAC-induced wall thinning will be monitored and addressed.

The NRC staff has reviewed the effects of the proposed MUR power uprate on FAC and the adequacy of the licensee's FAC program to predict the rate of material loss so that repair or replacement of affected components can be made before reaching a critical thickness. The staff finds the basis for the licensee's FAC program acceptable because it follows the recommendations of GL 89-08.

The NRC staff also reviewed the previous staff evaluations of the PBAPS FAC monitoring program that were conducted during the license renewal, EPU, and MELLLA+ reviews. During these previous evaluations, the staff found that the licensee's FAC monitoring program was acceptable and will accurately predict the rate of wall thinning for components susceptible to FAC.

For certain systems, the licensee stated that for predicted wall thinning, there are negligible changes in wear rate from the EPU evaluation. In addition, the licensee stated that no change to the piping inspection scope or frequency will be required in order to ensure adequate margin at MUR power uprate conditions.

The NRC staff has reviewed the licensee's evaluation of the proposed MUR power uprate on the FAC analysis and concludes that the licensee has adequately addressed the impact of changes in plant operating conditions on the FAC analysis. Additionally, the NRC staff concludes that the licensee has demonstrated that the updated analyses will predict, with reasonable assurance, the loss of material by FAC, and will ensure timely repair or replacement of affected components following implementation of the proposed MUR power uprate. The NRC staff has found that the FAC program will provide reasonable assurance that components susceptible to FAC will be managed appropriately post-MUR power uprate implementation. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to the impacts of FAC.

### 3.3.6 TSAR Section 3.6 – Reactor Recirculation System

The primary function of the reactor recirculation system is to vary the core flow and power during normal operation. The recirculation system also forms part of the reactor coolant system pressure boundary.

As discussed in TSAR Section 3.6, the licensee performed a plant-specific evaluation for the PBAPS reactor recirculation system using the evaluation approach presented in TLTR Section 5.6.2. The TPO uprate has a minor effect on the system and its components. Operation at the TPO uprated power is accomplished along an extension of the current MELLLA+ boundary on the power/flow map with no increase in the maximum core flow. No significant reduction of the maximum flow capability occurs due to the TPO uprate because of the small increase in core pressure drop of 0.01 pounds per square inch differential (psid). The effect on pump net positive suction head (NPSH) at TPO conditions is negligible. An evaluation confirmed that no significant increase in reactor recirculation system vibration occurs from the TPO operating conditions.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the reactor recirculation system from performing its intended functions.

### 3.3.7 TSAR Section 3.7 – Main Steam Line Flow Restrictors

As discussed in TSAR Section 3.7, in support of the EPU, the licensee previously performed a plant-specific evaluation of the main steam line (MSL) flow restrictors at 102 percent of CLTP using the approach described in TLTR Appendix J.2.3.7. A plant-specific evaluation of the effects of TPO RTP compared to CLTP determined that there is no change in operating temperature and no change in maximum operating dome pressure. Therefore, the resulting break flow rate remains unchanged. A slight decrease in operating pressure occurs along the steam line due to the higher flow rate pressure drop and less than 2 percent change in normal steam flow.

The plant-specific evaluation concluded that there is no increase in the steam flow for a main steam line break (MSLB) accident because the flow restrictor and operating pressure remain unchanged. Because the flow restrictors were designed and analyzed for the choke flow condition with the maximum pressure difference, which is bounding for the TPO uprate condition, the structural integrity of the MSL flow restrictors is not affected by a TPO uprate. Further, the less than 2 percent change in normal steam flow does not affect any accident-related loads because the current loads continue to bound the analysis for TPO uprate operation.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the MSL flow restrictors from performing their intended functions.

The NRC staff agrees with the licensee's conclusion that the requirements for the MSL flow restrictors remain unchanged for TPO uprate conditions and that all safety and operational aspects of the MSL flow restrictors are within previous evaluations.

### 3.3.8 TSAR Section 3.8 – Main Steam Isolation Valves

As discussed in TSAR Section 3.8 in support of the EPU, the licensee previously performed a plant-specific evaluation of the main steam isolation valves (MSIVs) at 102 percent of CLTP using the approach described in TLTR Appendix J.2.3.7. A plant-specific evaluation of the effects of TPO RTP compared to CLTP determined that there is no change in operating temperature, a slight decrease in operating pressure along the steam line due to the higher flow rate pressure drop, and less than 2 percent change in normal steam flow.

The plant-specific evaluation also concluded that there is no increase in the steam flow for an MSLB accident because the flow restrictor and operating pressure remain unchanged, and the less than 2 percent change in normal steam flow does not affect any accident-related loads because the current loads continue to bound the analysis for TPO uprate operation.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the MSIVs from performing their intended functions.

### 3.3.9 TSAR Section 3.9 – Reactor Core Isolation Cooling

The reactor core isolation cooling (RCIC) system provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for starting a low-pressure core cooling system. The RCIC system is designed to provide rated flow over a range of reactor pressures.

As discussed in TSAR Section 3.9, the licensee's plant-specific evaluation determined that there is no change in operating pressure, pressure setpoints of the SRVs, capability of the turbine-driven RCIC system to successfully develop the horsepower and speed required by the pumps, and RCIC capacity.

The plant-specific evaluation further concluded that the loss of feedwater (LOFW) analysis of record, including decay heat inputs, which was performed at 102 percent of CLTP, bounds the TPO uprate operating conditions, and the capability to maintain the water level above the top of active fuel (TAF) remains unchanged.

The licensee further stated that the conclusion in the LOFW analysis of record based on SAFER/GSTRM will remain valid with SAFER/PRIME, as the water level response between the SAFER/GSTRM and the SAFER/PRIME methodologies is expected to be essentially the same. The NRC staff's understanding is that the currently approved thermal conductivity degradation (TCD) model is incorporated in the PRIME code, not in the GSTRM code, and that degraded fuel thermal conductivity may result in higher fuel stored energy. This additional stored energy as an initial condition in the fuel may lead to a higher boil-off rate, resulting in a reduced water level in the core during an LOFW event. Therefore, the staff requested that the licensee explain why TCD is expected to have no impact on the calculation of water level during an LOFW event. The licensee's response in the supplement dated August 8, 2017, stated that during the LOFW event, which includes a reactor scram at the beginning of the event, the predicted fuel temperatures in the core would be higher with a fuel thermal model that accounts for TCD. This will affect the stored energy in the core at the start of the event. However, the stored energy dissipates over the period of time that the reactor water level falls to the low-level setting, which initiates the RCIC system. At the time of the RCIC system initiation (approximately 68 seconds after the reactor scram), the predominant source of energy to the coolant is decay heat, which is unaffected by the initial stored energy and TCD. The licensee, therefore, concluded that for the LOFW event analysis, the effect of stored energy (due to TCD) is small and insignificant, considering the available margin to top of active fuel during the LOFW event analysis. Based on the above discussion, the staff finds that the licensee's explanation is acceptable.

The minimum water level during an LOFW event is maintained at least 129 inches above TAF in the analysis of record. The TPO uprate does not affect the RCIC system operation, initiation, or capability requirements. In addition, the proposed power uprate does not increase the steady-state operating pressure or the SRV actuation setpoints. Based on these considerations, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the RCIC system from performing its intended functions.

### 3.3.10 <u>TSAR Section 3.10 – Residual Heat Removal System</u>

The residual heat removal system (RHR) is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal after reactor shutdown for both normal and post-accident conditions. The RHR system is designed to function in several operating modes. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, the shutdown cooling mode, the containment cooling mode, and fuel pool cooling assist mode. As discussed in TSAR Section 3.10, plant-specific evaluations were performed for PBAPS using the evaluation approaches provided in TLTR Section 5.6.4 and Appendices J.2.3.1 and J.2.3.13.

The licensee stated that the slightly higher decay heat has a small effect on the operation of the RHR system in the shutdown cooling mode. The ability of the RHR system to perform its required safety functions was demonstrated with analyses based on 102 percent of CLTP. Therefore, all safety aspects of the RHR system are within previous evaluations.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the RHR system from performing its intended functions.

### 3.3.11 TSAR Section 3.11 – Reactor Water Cleanup System

As discussed in UFSAR Section 4.9.1, the function of the reactor water cleanup (RWCU) system is to maintain high reactor water purity to limit chemical and corrosive action, thereby limiting fouling and deposition on heat transfer surfaces. The RWCU system also removes corrosion products to limit impurities available for neutron activation and resultant radiation from deposition of corrosion products.

As discussed in TSAR Section 3.11, the licensee performed a plant-specific evaluation of the reactor water cleanup (RWCU) system at PBAPS using the evaluation approach provided in Section 5.6.6 and Appendix J.2.3.4 of the TLTR. This plant-specific evaluation verified that there would be no change in the nominal operating pressure and no significant change in nominal operating temperature (less than 1 °F) in the high-pressure portion of the system for the PBAPS TPO uprate. The evaluation also verified there would be no identifiable change in level of impurities in the reactor water and that the capacity of the RWCU system would be sufficient, possibly with small operational adjustments, to accommodate the small effect of the TPO uprate on RWCU duty.

The NRC concludes that based on the plant-specific evaluation, there is reasonable assurance that the TPO will not impact the capability of the RWCU system to perform its intended functions.

## 3.4 TSAR Section 4.0 – Engineered Safety Features

The following provides the NRC staff's technical review of the topics in Section 4.0 of the TSAR.

### 3.4.1 TSAR Section 4.1 – Containment System Performance

Appendix G to the TLTR outlines the methods, approach, and scope of plant-specific containment analyses, which have been used by the licensee in support of the TPO uprate. The TLTR states that the previous LOCA containment analyses are bounding for the TPO uprate because they considered 2 percent uncertainty in the RTP as required by the previous methodology. Although the TPO uprate will increase the nominal operating conditions slightly, the required bounding conditions for the limiting analytical cases do not change from the previously documented bounding conditions. The TLTR states that there is no effect of the TPO uprate on the containment pressure and temperature response and dynamic loads due to LOCA and safety relief valve (SRV) actuation.

As discussed in TSAR Section 4.1, the following current licensing basis containment analyses were performed at 102 percent of the CLTP power level of 3,951 MWt (i.e., at 4,030 MWt), which bounds the TPO power level (i.e., 4,016 MWt):

- Short-term response for peak containment pressure
- Short-term response for drywell gas temperature
- Long-term suppression pool temperature response for bulk pool temperature
- Long-term suppression pool temperature response for local temperature with SRV discharge
- Subcompartment pressurization loads
- LOCA dynamic loads
- SRV discharge loads

The licensee stated that although the nominal operating conditions change slightly because of the TPO uprate, the initial conditions and inputs remain the same as in the CLTP analysis, which supported the EPU license amendment (Reference 4) and the MELLLA+ license amendment (Reference 7).

As discussed in the NRC staff's SE for the EPU, the licensee used NRC-accepted computer codes LAMB and M3CPT for the above-mentioned current short-term analyses. The current long-term suppression pool temperature response was performed using the NRC-accepted GE computer codes LAMB and Super-Hex. The decay heat model was based on American Nuclear Society (ANS) 5.1- 1971 plus 20 percent for short-term analysis, and ANS 5.1-1979 plus 2 sigma for long-term analysis.

The NRC staff concludes that the above analyses are not affected for the TPO power uprate because they were performed at 102 percent of the CLTP power level, which bounds the proposed TPO power level.

In addition to the analyses listed above, as discussed in the licensee's supplement dated August 8, 2017, the following current licensing basis containment analyses were performed at 102 percent of the CLTP power level:

- Most limiting containment temperature response for equipment environmental qualification
- Peak containment wall temperature for structural analysis

The NRC staff concludes that the above analyses are not affected for the TPO power uprate because they were performed at 102 percent of the CLTP power level, which bounds the proposed TPO power level.

*TSAR Section 4.1.1 – GL 89-10*

NRC GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" (Reference 32), extended the scope of the program outlined in IE Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," and its Supplement 1, to include all safety-related motor-operated valves (MOVs), as well as all position-changeable MOVs. The licensees were requested to develop and implement a program to ensure that the switch settings (torque, torque bypass, position limit, and overload) on the safety-related MOVs are selected, set, and maintained correctly to accommodate the maximum differential pressures expected on these valves during both normal and abnormal events within the design basis during the life of the plant.

As discussed in TSAR Section 4.1.1, the licensee stated that the current analyses of safety-related MOVs within the GL 89-10 program use maximum line pressures, maximum differential pressures, and maximum ambient temperatures that bound the operation at TPO conditions.

The NRC staff considers the licensee's evaluation of the GL 89-10 MOVs acceptable because the current analyses bound the operating conditions at the TPO uprate, and no changes are necessary to meet the functional requirements of these MOVs at the TPO uprate conditions.

*TSAR Section 4.1.2 – GL 96-05*

NRC GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves" (Reference 33), discusses the periodic verification of the capability of safety-related MOVs to perform their safety functions consistent with the current licensing bases of nuclear power plants. This GL provides more complete guidance regarding periodic verification of safety-related MOVs and supersedes GL 89-10 and its supplements with regard to MOV periodic verification.

As discussed in TSAR Section 4.1.2, the licensee stated that the current evaluation of the GL 96-05 program was reviewed and determined to have no effects related to the TPO uprate. In the supplement dated August 8, 2017, the licensee provided the following additional information:

> Safety related Motor Operated Valves (MOVs) in the GL 96-05 Program are contained in the following systems: Main Steam (MS), Reactor Recirculation (RR), Feedwater (FW), Reactor Water Cleanup (RWCU), Residual Heat Removal (RHR), Reactor Core Isolation Cooling (RCIC), Core Spray (CS), High Pressure Coolant Injection (HPCI), High Pressure Service Water (HPSW), Emergency Service Water (ESW), Reactor Building Closed Cooling Water (RBCCW), Drywell Cooling, and Emergency Cooling Tower (ECT).
>
> - Reactor pressure is not changed by TPO, therefore analyzed MOV conditions are not impacted for MS, RR and RWCU systems.
> - MOVs in the RHR, RCIC, CS, HPCI and RBCCW systems are evaluated for post-accident conditions which are analyzed at 102% of CLTP which bounds TPO. Additionally, RCIC and HPCI system valves are analyzed and found to be acceptable for increased peak reactor pressure under ATWS conditions at TPO.
> - TPO has no effect on the HPSW, ESW and ECT systems' maximum motor ambient temperatures, maximum line pressures or differential pressures. Therefore, analyzed MOV conditions are not impacted for HPSW, ESW, and ECT systems.
> - The only GL 96-05 Program MOVs in the FW system are the startup recirculation isolation valves, which have a safety function to close only during startup and shutdown and are normally closed at full power operation. Operating conditions during startup and shutdown are not impacted by TPO.

- The GL 96-05 Program MOVs in the Drywell Cooling system are evaluated for post-LOCA drywell pressure, which is analyzed at 102% of CLTP and therefore bounds TPO. The maximum analyzed ambient temperature is based on a ruptured main steam line. As the main steam conditions are not changed by TPO, the analyzed MOV ambient temperature is not changed by TPO.

Based on the above, the NRC staff agrees with the licensee's justification that the GL 96-05 valves would not be affected by the TPO uprate.

*TSAR Section 4.1.3 – GL 95-07*

NRC GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves" (Reference 34), requests that licensees perform or confirm that it previously performed: (1) evaluations of operational configurations of safety-related, power-operated (including motor-, air-, and hydraulically-operated) gate valves for susceptibility to pressure locking and thermal binding; and (2) further analyses, and any needed corrective actions, to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing the safety functions within the current licensing bases of the facility.

In TSAR Section 4.1.3, the licensee stated that the criteria for susceptibility to pressure locking or thermal binding were reviewed at the TPO uprate and it was determined that the expected slight changes in operating or environmental conditions from the TPO uprate would have no effect on the functioning of power-operated gate valves within the scope of GL 95-07, and that the valves would remain capable of performing their design-basis functions.

The NRC staff agrees with the licensee's evaluation with respect to GL 95-07 at the TPO uprate conditions because there is reasonable assurance that minor changes in the operating or environmental conditions would not affect the operation of the safety-related power-operated gate valves.

*TSAR Section 4.1.4 – GL 96-06*

NRC GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions" (Reference 35), identifies the following potential problems with equipment operability and containment integrity during DBA conditions: (1) cooling water systems serving the containment air coolers may be exposed to water hammer during postulated accident conditions; (2) cooling water systems serving the containment air coolers may experience two-phase flow conditions during postulated accident conditions; and (3) thermally-induced over-pressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity by bypass leakage. GL 96-06 questioned whether the higher heat loads at accident conditions could potentially cause steam bubbles, water hammer, and two-phase flow due to the higher outlet temperatures from cooled components, particularly the containment fan coolers.

As discussed in TSAR Section 4.1.4, the licensee stated that the current evaluation of the issues identified in GL 96-06 do not change for the TPO uprate. In the supplement dated August 8, 2017, the licensee further stated that to support the EPU, the post-accident drywell conditions were analyzed at 102 percent of CLTP, which bounds the TPO uprate.

The NRC staff finds the licensee's justification acceptable because the current licensing basis analysis was performed at 102 percent CLTP, which bounds the TPO power uprate.

*TSAR Section 4.1.5 – GL 89-16*

NRC GL 89-16, "Installation of a Hardened Wetwell Vent" (Reference 36), discusses the advantages of installing a hardened containment (wetwell) vent and requested information from licensees on installation of such a vent. This was a result of the NRC's BWR Mark I Containment Performance Improvement Program.

As discussed in TSAR Section 4.1.5, the licensee indicated that the licensing basis with respect GL 89-16 was reviewed and determined to not be impacted by to the TPO uprate. In the supplement dated August 8, 2017, the licensee further stated that the required relieving capacity of the hardened wetwell vent was analyzed at 102 percent of CLTP, which bounds TPO conditions.

The NRC staff finds the licensee's justification acceptable because the current licensing basis analysis was performed at 102 percent CLTP, which bounds the TPO power uprate.

*TSAR Section 4.1.6 – Containment Coatings*

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff reviewed the protective coating systems used inside containment for their suitability for and stability under design-basis loss-of-coolant accident (DBLOCA) conditions considering temperature, pressure, radiation, and chemical effects. The NRC's acceptance criteria for protective coating systems are based on: (1) 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," which covers quality assurance requirements for the design, fabrication, and construction of safety-related SSCs; and (2) RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Revision 2, dated October 2010 (ADAMS Accession No. ML102230344), which covers application and performance monitoring of coatings in nuclear power plants.

In TSAR Sections 4.1 and 4.1.6, the licensee stated that the nominal operating conditions change slightly, but the initial conditions for containment analysis do not change from the current licensing basis (i.e., EPU and MELLLA+). In addition, the licensee stated that the maximum post-accident containment conditions do not change. The licensee also stated that the service level 1 coatings are qualified to 340 °F, 70 psi, and $\geq 1 \times 10^9$ rads. The licensee concluded that the containment coatings continue to bound DBA temperature, pressure, and radiation at the TPO conditions. Because the post-DBLOCA conditions in containment do not change, the coating qualifications continue to bound the predicted conditions in containment after a DBLOCA at the proposed TPO conditions. Therefore, the NRC staff has reasonable assurance that the coatings will not be adversely impacted by the power uprate conditions and finds the MUR power uprate acceptable with respect to protective coatings.

### 3.4.2 TSAR Section 4.2 – Emergency Core Cooling Systems

The emergency core cooling systems (ECCS) are designed to provide protection in the event of a LOCA due to a rupture of the primary system piping. Although DBAs are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (1) the peak cladding temperature, (2) local cladding oxidation, (3) total hydrogen generation, (4) coolable core geometry, and (5) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analyses identify the break sizes that most severely challenge the ECCS systems and the primary containment. The maximum average planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA analysis. The licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for PBAPS includes the high-pressure coolant injection (HPCI) system, the low-pressure coolant injection (LPCI) mode of the RHR system, the low-pressure core spray (CS) system, and the automatic depressurization system (ADS). Each of these systems is described below.

*TSAR Section 4.2.1 – High Pressure Coolant Injection*

The HPCI system is a steam turbine-driven system designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small-break LOCA that does not immediately depressurize the RPV.

As discussed in TSAR Section 4.2.1, for the TPO uprate, there is no change to the nominal reactor operating pressure or the SRV setpoints. A plant-specific evaluation for PBAPS confirmed that there is no change in the operating pressure, pressure setpoints of the SRVs, capability of the turbine-driven HPCI system to successfully develop the horsepower and speed required by the pumps, startup capability of the turbine startup logic, and HPCI capacity. The TPO uprate does not affect the HPCI system operation, initiation, or capability requirements.

In response to an NRC staff question, the licensee confirmed in the supplement dated August 8, 2017, that the ability of the HPCI system to perform required safety functions is demonstrated with previous analyses based on 102 percent of CLTP. Therefore, all safety aspects of the HPCI system are within previous evaluations, and the requirements are unchanged for the TPO uprate conditions.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the HPCI system from performing its intended functions.

*TSAR Section 4.2.2 – Core Spray*

The core spray (CS) system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant makeup for a large-break LOCA and for any small-break LOCA after the RPV has depressurized. It also provides spray cooling for long-term core cooling in the event of a LOCA.

As discussed in TSAR Section 4.2.2, a plant-specific evaluation of the CS system was performed for PBAPS using the evaluation approach provided in TLTR Section 5.6.10 and Appendix J.2.3.1. The plant-specific evaluation determined that there is no change in CS capacity and decay heat removal capability. The TPO uprate does not affect the CS system operation, initiation, or capability requirements.

In response to an NRC staff question, the licensee confirmed in the supplement dated August 8, 2017, that the ability of the CS system to perform required safety functions is demonstrated with previous analyses based on 102 percent of CLTP. Therefore, all safety aspects of the CS system are within previous evaluations, and the requirements are unchanged for the TPO uprate conditions.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the CS system from performing its intended functions.

*TSAR Section 4.2.3 – Low Pressure Coolant Injection*

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to provide reactor vessel coolant makeup during a large-break LOCA or small-break LOCA after the RPV has depressurized.

As discussed in TSAR Section 4.2.3, a plant-specific evaluation of the LPCI mode was performed for PBAPS using the evaluation approach provided in TLTR Section 5.6.4. The plant-specific evaluation determined that there is no change in LPCI mode capacity and decay heat removal capability. The TPO uprate does not affect the LPCI mode of the RHR system operation, initiation, or capability requirements.

In response to an NRC staff question, the licensee confirmed in the supplement dated August 8, 2017, that the ability of the LPCI system to perform required safety functions is demonstrated with previous analyses based on 102 percent of CLTP. Therefore, all safety aspects of the LPCI system are within previous evaluations, and the requirements are unchanged for the TPO uprate conditions.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the LPCI system from performing its intended functions.

*TSAR Section 4.2.4 – Automatic Depressurization System*

The ADS uses the SRVs to reduce reactor pressure after a small-break LOCA with high-pressure systems failure, LPCI, and CS to provide cooling flow to the vessel. The plant design requires SRVs to have a minimum flow capacity. After a delay, the ADS actuates either on low water level plus high drywell pressure or on low water level alone.

As discussed in TSAR Section 4.2.4, a plant-specific evaluation determined that the pressure setpoints of the ADS valves are unchanged, and the ADS initiation logic and the ADS valve control are not affected by the TPO uprate operating conditions. The plant-specific evaluation also concluded that the performance of the existing ADS valves remains unchanged because the current small-break LOCA analysis performed at 102 percent of CLTP bounds the TPO uprate conditions. The TPO uprate does not affect the ADS system operation, initiation, or capability requirements.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the changes associated with the TPO uprate will not impact the capability of the ADS system from performing its intended functions.

*TSAR Section 4.2.5 – Emergency Core Cooling System Net Position Suction Head*

The net positive suction head (NPSH) for the ECCS pumps that draw water from the suppression pool during a design-basis accident or a special event depends on the suppression pool temperature response. For the current licensing basis containment analyses, the suppression pool temperature was analyzed at 102 percent CLTP for the recirculation suction line break and small steam line break LOCAs. Therefore, the results for these analyses are bounding for the TPO power level.

For the non-design-basis events (also called special events), which are Appendix R fire, station blackout (SBO), and ATWS, the current licensing basis analysis was performed at 100 percent CLTP. As discussed in TSAR Section 4.2.5, for the TPO uprate, the licensee analyzed these events at the TPO bounding thermal power of 101.7 percent of CLTP (i.e., 4,018 MWt). The licensee used the same methodology for containment analysis as in the EPU analysis and, accordingly, did not credit containment accident pressure (CAP) in the net positive suction head (NPSH) analysis. The licensee stated that for the Appendix R fire event, the EPU analysis conservatively used an emergency service water (ESW) temperature of 92 °F instead of its nominal value of 86 °F, as allowed by the draft guidance in SECY-11-0014, Enclosure 1 (Reference 37). The nominal value of 86 °F is based on a statistical analysis of a 5-year sampling of ESW temperature data for the months of June, July, August, and September. The licensee removed the conservatism and provided revised analysis results of the NPSH Available (NPSHA) and NPSH margins for the Appendix R fire cases A1, C1A, and C1B in Tables 4-1 and 4-2 of the TSAR. The revised analysis resulted in a higher NPSH margin for the CLTP (i.e., EPU) Appendix R fire cases.

In an RAI, the NRC staff requested the licensee to justify the change in any of the remaining input parameters besides the core thermal power and ESW temperature that reduced conservatism in the suppression pool temperature response. In the supplement dated August 8, 2017, the licensee stated that the TPO uprate evaluation is based on the Appendix R fire event key input parameters listed in Table 2.5-1 of Attachment 6 to the EPU LAR application dated September 28, 2012 (Reference 38). The following table provides the revised CLTP (EPU) and the TPO values of NPSH margins for the special events. The NPSH margin for the EPU Appendix R fire event in the table represents the revised margin for the limiting Case C1B for the RHR pump.

NPSH Margins for Special Events

| Special Event | NPSH Margin (feet) | |
|---|---|---|
| | **CLTP (EPU)** | **TPO** |
| Appendix R Fire (Note 1) | 0.88 (Note 2) | 0.28 |
| SBO (Note 3) | 4.75 | 4.18 |
| ATWS (Note 4) | 14.76 | 14.57 |
| Note 1: Limiting Case C1B, RHR pump per TSAR Table 4-2<br>Note 2: Revised value<br>Note 3: SBO values per TSAR Table 4-4<br>Note 4: ATWS values per TSAR Table 4-3 | | |

The licensee's TPO uprate analysis for special events results for the time in the zone of maximum erosion (i.e., the transient time below NPSH margin ratio of 1.6) meets the draft guidance in SECY-11-0014, Enclosure 1.

As noted in the above table, the decrease in the NPSH margin for the TPO uprate condition from that of EPU is due to a higher suppression pool temperature at the TPO condition. However, the analysis for these events shows positive NPSH margin without crediting CAP and an acceptable time of operation in the zone of maximum erosion.

As discussed above, the NRC staff concludes that the licensee's TPO NPSH evaluation for the design-basis LOCAs for the ECCS pumps is acceptable because the TPO uprate conditions are bounded by the current analysis performed at 102 percent CLTP. Also, the NRC staff concludes that the TPO NPSH evaluation for special events, based on analyses performed at 101.7 percent of CLTP, is acceptable because the licensee used the same methodology as in the current analysis and demonstrated positive NPSH margin and acceptable transient time in the maximum erosion zone for ECCS pumps. The licensee's TPO analysis for special events at 101.7 percent CLTP is conservative because it bounds the TPO uprate power of 101.66 percent of the CLTP.

*TSAR Section 4.2.5.1 – ECCS Suction Strainer Debris Loading*

As discussed in TSAR Section 4.2.5.1, the licensee's NPSH evaluation, at TPO conditions, included consideration for ECCS suction strainer debris loading. Specifically, suction strainer debris loading was considered for LOCAs, small steam line breaks, and ATWS events.

3.4.3 TSAR Section 4.3 – Emergency Core Cooling System Performance

The ECCS is designed to provide protection against a postulated LOCA caused by ruptures in the primary system piping. As discussed in TSAR Section 4.3, the current LOCA analyses for PBAPS, which consider operation under EPU and MELLLA+ conditions, was performed at 102 percent of CLTP and, therefore, bound TPO uprate conditions, consistent with Appendix K to 10 CFR Part 50. The ECCS-LOCA results for PBAPS are in conformance with the licensing requirements of 10 CFR 50.46. Therefore, the CLTP LOCA analysis for GNF2 fuel bounds the TPO uprate for PBAPS.

As shown in Table 16.1-1 of Reference 44, the PBAPS LOCA analyses resulted in a licensing basis peak cladding temperature (PCT) of less than 1920 °F for GNF2 fuel, maximum cladding oxidation less than 4.0 percent, and maximum hydrogen generation less than 0.1 percent. These results comply with the 10 CFR 50.46 requirements of PCT of less than 2200 °F, maximum cladding oxidation less than 17 percent, and maximum hydrogen generation less than 1 percent. The licensee's analyses are based on the NRC-approved methodology described in GESTAR II and on bounding power and flow conditions.

Furthermore, the LOCA analyses of record demonstrate that the HPCI system, the LPCI mode of RHR, the CS system, and the ADS have the capabilities to provide core cooling during a LOCA. These capabilities do not change for operation at the uprated conditions. The ECCS will, therefore, continue to meet the ECCS-LOCA analysis assumptions and design criteria at the uprated conditions.

Based on the considerations discussed above, the NRC staff concludes there is reasonable assurance that the ECCS will perform as designed and analyzed at the uprated conditions.

3.4.4 <u>TSAR Section 4.4 – Main Control Room Atmosphere Control System</u>

As discussed in Appendix J, Section J.2.3.8, of the TLTR, the main control room atmosphere control system was considered from the viewpoint of continued assurance of habitability of the control room following a postulated accident at TPO uprate conditions. The TLTR states that compliance in this area is unchanged if the system had previously been evaluated for accident conditions at 102 percent of CLTP.

TSAR Section 4.4 confirmed that the control envelope/habitability systems had previously been evaluated for radiation release accident conditions at 102 percent of CLTP. Accordingly, the NRC staff concludes that TPO uprate will not impact the capability of the system from performing its safety function at TPO conditions.

3.4.5 <u>TSAR Section 4.5 – Standby Gas Treatment System</u>

As discussed in Appendix J, Section J.2.3.9, of the TLTR, the standby gas treatment system (SGTS) is designed to minimize offsite and control room dose rates during venting and purging of the containment atmosphere. The SGTS maintains the secondary containment at a slightly negative pressure during such conditions. The TLTR states that that the capability of the SGTS is unchanged at TPO conditions if the system had previously been evaluated for accident conditions at 102 percent of CLTP.

TSAR Section 4.5 confirmed that the SGTS can currently accommodate DBA conditions at 102 percent of CLTP. Accordingly, the NRC staff concludes that TPO uprate will not impact the capability of the system from performing its safety function at TPO conditions.

3.4.6 TSAR Section 4.6 – Primary Containment Leak Rate Test Program and Containment Isolation System

Section 4.6 of the TSAR addressed the primary containment leak rate test (PCLRT) program and the containment isolation system. The TSAR included the following statements:

> The PCLRT program is not affected, because the reactor operating parameters are not changed for the TPO uprate and the current containment response analyses have been performed at 102% of CLTP. Based on no change in the post-accident short-term containment pressure and temperature, there is no revision necessary to the 10 CFR 50 Appendix J testing methodology and/or acceptance test criteria.
>
> The containment isolation system is not affected by TPO uprate. The system uses setpoints developed to ensure containment isolation based on postulated accidents as expressed in the UFSAR considering the current licensing basis which includes approved amendments for EPU [Reference 4] and MELLLA+ [Reference 7]. These setpoints utilize a 2% uncertainty factor required by RG 1.49. Because the TPO uprate reduces the RG 1.49 uncertainty from 2% to 0.34%, the previous analysis remains bounding.

Since the current analyses performed for 102 percent of CLTP bound operation at TPO conditions, the NRC staff concludes that the TPO uprate is acceptable with respect to the PCLRT program and the containment isolation system.

3.4.7 TSAR Section 4.7 – Post-LOCA Combustible Gas Control System

The PBAPS, Units 2 and 3, containments are inerted with nitrogen. The NRC revised 10 CFR 50.44, "Combustible gas control for nuclear power reactors," on September 16, 2003. The changes eliminated the requirements for hydrogen recombiners for nitrogen inerted containments and relaxed the requirements for hydrogen and oxygen monitoring in containment. The revised regulation no longer defines a design-basis LOCA hydrogen release and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. License Amendment Nos. 256 and 259 for PBAPS, Units 2 and 3, respectively, issued on August 11, 2005 (Reference 41), eliminated the requirements for the hydrogen and oxygen monitors. License Amendment Nos. 274 and 278 for PBAPS, Units 2 and 3, respectively, issued on January 28, 2010 (Reference 42), eliminated the requirements for the hydrogen recombiners (i.e., Containment Atmospheric Dilution system). Therefore operating under TPO conditions does not affect the current combustible gas control system.

3.5 TSAR Section 5.0 – Instrumentation and Control

SE Sections 3.5.1 through 3.5.3 provide the NRC staff's technical review of the topics in Section 5.0 of the TSAR. In addition, SE Section 3.5.4 addresses the staff's review of the thermal power measurement uncertainty associated with the LEFM system.

3.5.1 TSAR Section 5.1 – NSSS Monitoring and Control

Section 5.1 of the TSAR states that instrumentation and controls that directly interact with or control the reactor are usually considered within the nuclear steam supply system (NSSS). The NSSS monitoring and control systems evaluated in TSAR Section 5.1 are discussed below.

*TSAR Section 5.1.1.1 – Average Power Range Monitors and Wide Range Neutron Monitors*

As discussed in the PBAPS UFSAR Section 7.5.7 and the PBAPS TS Bases, the average range power monitor (APRM) channels provide the primary indication of neutron flux within the core. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRMs are capable of generating a scram trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage. To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance.

As discussed in TSAR Section 5.1.1.1, the APRMs will be recalibrated to indicate 100 percent at the TPO RTP level of 4,016 MWt. The NRC concludes that this change is acceptable since it is necessary to rescale the APRM signals consistent with the new power level.

As discussed in UFSAR Section 1.6.4.1.3, the wide range neutron monitors provide flux level indication during reactor startup and low power operation. As discussed in TSAR Section 5.1.1.1, no adjustment is needed to ensure the wide range neutron monitors have adequate overlap with the APRMs. This is consistent with the disposition in Section 5.6.1 of the TLTR and is, therefore, acceptable.

*TSAR Section 5.1.1.2 – Local Power Range Monitors and Traversing In-Core Probes*

As discussed in TSAR Section 5.1.1.2, in accordance with Methods LTR Limitation and Condition 9.17 and the M+ LTR Limitation and Condition 12.15, the predicted bypass void fraction at the D-level LPRMs satisfies the [[ ]] design requirement for MELLLA+ and TPO conditions at PBAPS. The supplemental reload licensing report will validate that the power distribution in the core is achieved while maintaining individual fuel bundles within the allowable thermal limits as defined in the Core Operating Limits Report. While moving down and left on the MELLLA+ upper boundary, the hot channel exit void in the bypass region increases. The hot channel exit void in the bypass region does not exceed [[ ]] in the MELLLA+ operating domain, including the expanded TPO region, as shown in Table 5-1 of the TSAR and, therefore, meets the limitations and conditions.

Because thermal neutron traversing in-core probes (TIPs) are affected by bypass voiding above the D-level LPRMs in excess of [[ ]], operator actions and procedures that mitigate the effect of bypass voiding on the thermal TIPs and the core simulator used to monitor the fuel performance are requested in M+ LTR Limitation and Condition 12.15 for operation. However, these requirements are not applicable for PBAPS because of the use of gamma TIPs and because hot channel bypass voiding at the TIP exit elevation is not in excess of [[ ]] for the entire MELLLA+ operating domain, including the expanded TPO region. Based on these considerations, the NRC staff finds this acceptable.

*TSAR Section 5.1.1.3 – Rod Block Monitor*

As discussed in UFSAR Sections 1.6.4.1.3 and 7.5.8, the rod block monitor (RBM) instrumentation is provided to prevent rod withdrawal when reactor power should not be increased. The RBM system has two RBM channels, each of which uses input signals from a number of LPRM channels.

As discussed in TSAR Section 5.1.1.3, the RBM instrumentation is not significantly affected by the TPO uprate conditions, and no change is needed. The NRC staff finds there is reasonable assurance the RBM system will operate as intended since the APRMs will be rescaled to the new power level.

*TSAR Section 5.1.2 – Rod Worth Minimizer*

As discussed in UFSAR Section 7.16.3.3, the rod worth minimizer (RWM) function assists and supplements the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low power level control rod procedures.

As discussed in the PBAPS TS Bases, control rod patterns during startup conditions are controlled by the operator and the RWM (LCO 3.3.2.1, "Control Rod Block Instrumentation") so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10 percent RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a control rod drop accident.

The RWM low power setpoint is unchanged in terms of percent power for TPO. This setpoint defines the power level below which the RWM is required. Maintaining this function in effect until 10 percent RTP will result in a larger RWM range in terms of absolute power. Therefore, the NRC staff concludes that not revising this setpoint is conservative for TPO.

3.5.2 <u>TSAR Section 5.2 – BOP Monitoring and Control</u>

Section 5.2 of the TSAR addresses various control and protective systems that may be affected by changes in operating pressures and temperatures. These systems include the reactor pressure control system (PCS), the main turbine electro-hydraulic control (EHC) system, the FW control system, and various leak detection systems as discussed below. The control systems are not safety-related, but excessive challenges to the systems may increase the frequency of transients. The leak detection systems provide a protective function, but if actuated, also introduce plant transients.

*TSAR Section 5.2.1 – Pressure Control System*

Section 5.2.1 of the TSAR included a description of the reactor PCS. The PCS controls reactor pressure by modulating turbine control valve position and through use of the turbine bypass system. The analysis presented in the TSAR indicated that no modification to the turbine bypass system was necessary, but the turbine control valve position necessary for operation at the full TPO uprate power may exceed the procedural limit on turbine control valve position. The procedural limit provides adequate throttling margin for pressure control, and that limit would be maintained at TPO uprate conditions. The TSAR stated that turbine-related hardware may be modified to take full advantage of the TPO power uprate, and such modifications would be implemented under the provisions of 10 CFR 50.59. Therefore, adequate reactor pressure control would be maintained at TPO uprate conditions.

*TSAR Section, 5.2.2 – EHC Turbine Control System and Section 5.2.3 – Feedwater Control System*

Sections 5.2.2 and 5.2.3 of the TSAR addressed the operation of the EHC and FW control systems, respectively. The analyses presented in the TSAR indicated that these systems had adequate control margin to accommodate the small changes in flow associated with operation at TPO uprate conditions. Therefore, adequate EHC and FW control system performance would be maintained at TPO uprate conditions.

*TSAR Section 5.2.4 – Leak Detection System*

Section 5.2.4 of the TSAR discussed the effect of the TPO on leak detection system margin. For operation at TPO uprate conditions, the nominal reactor vessel steam dome pressure and temperature would be unchanged, and the FW temperature would increase by only a small amount (~2 °F). The majority of leak detection system setpoints consider the maximum vessel temperature in the steam dome for flow through the system, and are, therefore, unaffected by the TPO uprate. The FW temperature increase results in a negligible increase in normal temperatures within the steam tunnel and RWCU areas. Therefore, adequate margins to actuation of the leak detection systems on high area temperatures are maintained for operation at TPO conditions.

*TSAR Section 5.2 – Conclusion*

The NRC concludes that based on the licensee's evaluation provided in TSAR Section 5.2, there is reasonable assurance that the TPO uprate will not impact the ability of the balance-of-plant monitoring and control systems from performing their intended functions.

3.5.3 <u>TSAR Section 5.3 – Technical Specification Instrument Setpoints</u>

The NRC staff's review of the TS changes for this LAR is contained in SE Section 3.11.

3.5.4 <u>Thermal Power Measurement Uncertainty</u>

*Background – LEFM Technology and Measurement*

Nuclear power plants are licensed to operate at a specified core thermal power. Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 requires ECCS analyses to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed thermal power level (i.e., 102 percent RTP) to allow for instrumentation uncertainties. Alternatively, Appendix K allows such analyses to assume a value lower than 102 percent RTP (but not less than the licensed thermal power level), provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. This allowance gives licensees the option of justifying a power uprate with reduced margin between the power level assumed in the ECCS analysis and the licensed power level by using more accurate instrumentation to calculate the reactor thermal power.

Cameron LEFM systems, such as the LEFM system installed at PBAPS, Units 2 and 3, use transit time methodology to measure fluid velocity. The basis of the transit time methodology for measuring fluid velocity and temperature is that ultrasonic pulses transmitted through a fluid stream travel faster in the direction of the fluid flow than through the opposite flow. The difference in the upstream and downstream traversing times of the ultrasonic pulse is proportional to the fluid velocity in the pipe. The temperature is determined using a correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

Multiple diagonal acoustic paths are used in the LEFM, allowing velocities measured along each path to be numerically integrated over the pipe cross-section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross-section area, and the fluid density to determine the feedwater mass flow rate. The velocity profile correction factor is derived from calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

The LEFM system transducers at PBAPS are installed upstream of the original feedwater flow venturis and are physically located outside of the third and fourth stage feedwater heater rooms on elevation 135' in a designated locked high radiation area at power. The electronics cabinet is installed in the main lube oil equipment area in the turbine building on elevation 135', where the radiation field is generally less than 2 millirem per hour at full power.

The LEFM system consists of three LEFM flow meters, one in each of the three feedwater lines. Each flow meter contains flow transducers arranged in two planes. Plane 1 consists of flow transducer paths 1 through 4, and Plane 2 consists of flow transducer paths 5 through 8. The flow data from an LEFM flow meter with a single functioning plane has greater associated measurement uncertainty than that from a LEFM flow meter with both planes functioning, but less associated measurement uncertainty than that from a feedwater flow nozzle (i.e., venturi).

The LEFM system is used to provide feedwater flow input for the plant thermal heat balance calculation. The currently installed feedwater flow venturis will remain available for use if the LEFM is not functional. The Cameron LEFM flow meters have two operating modes (Normal and Maintenance) and a Fail mode as follows:

- CheckPlus Mode (Normal Mode): When in the CheckPlus mode, a system normal is displayed when all the feedwater flow, temperature, and header pressure signals for feedwater lines A and B are normal and operating within design limits. Calculated power level uncertainty associated with the LEFM flow measuring system in this condition is less than 0.34 percent. Per the LAR, the plant will be able to operate at up to 4,016 MWt when all three LEFM flow meters are in CheckPlus Mode.

- Check Mode (Maintenance Mode): The LEFM system Check mode indicates a loss of function that causes it to operate outside the specified accuracy of ± 0.34 percent. Typically, this occurs due to a malfunction of a single path or plane and results in an uncertainty increase to ± 0.5 percent. When the LEFM system shifts from the CheckPlus mode to the Check mode, a visual alarm indicates that there has been a loss of LEFM system redundancy. In the event of a failure of one path or plane, in any, or all of the three LEFM flow meters, that cannot be restored to full functionality within 72 hours, power will be reduced to ≤ 4,010 MWt.

- Fail Mode: The LEFM system's Fail mode indicates a loss of function that causes the LEFM system to operate outside the specified accuracy envelope of ± 0.5 percent. If any of the three LEFM flow meters are in Fail mode, the power level uncertainty reverts to the 2.0 percent associated with the venturi flow meters and power will be reduced to ≤ 3,951 MWt within 72 hours if LEFM functionality cannot be restored.

The LEFM system has continuous operating online self-diagnostics to verify the digital circuits are operating correctly and within the design-basis uncertainty limits. These diagnostics can identify failure conditions that will cause the LEFM flow meters to switch from the CheckPlus mode to the Check (Maintenance) or Fail modes. Validated LEFM data, including calculated results, status, and signal process information, is sent to the plant monitoring system (PMS) computer at regular intervals. The PMS provides a visual alarm, upon change in the status of an LEFM flow meter, on the operator overview display screen. In Section 3.3.4 of Attachment 1 to the application, the licensee stated, "[i]f the LEFM system or a portion of the system becomes inoperable, control room operators are promptly alerted by a plant monitoring system (PMS) alarm in the control room. Feedwater flow input to the core thermal power calculation would then be transferred to the feedwater flow nozzles in accordance with station procedures."

The following sections provide the NRC staff's review of the licensee's justification regarding the thermal power measurement uncertainty for the proposed power uprate at PBAPS, Units 2 and 3.

*Disposition of NRC Criteria for Use of LEFM Topical Reports*

As discussed in Section 3.3.1 of Attachment 1 to the licensee's application, the LEFM system flow measurement method is described in Caldon Topical Reports ER-80P (Reference 50) and ER-157P (Reference 51). The NRC staff's SEs approving these reports (Reference 52 and Reference 53) established nine criteria for use of these topical reports in plant-specific LARs. Section 3.3.4 of Attachment 1 to the licensee's application addressed these criteria as follows.

<u>*Criterion 1*</u>
*The licensee should discuss the maintenance and calibration procedures that will be implemen*ted with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

*Calibration and Maintenance*

The licensee stated that necessary procedures and documents required for maintenance and calibration of the LEFM system have been implemented to ensure that the system is properly maintained and calibrated. Preventive maintenance scope and frequency is based on vendor recommendations and performance data reviews. Transducers are replaced as determined to be necessary by a review of the equipment's performance history by the LEFM system vendor.

For instrumentation other than the LEFM system that contributes to the power calorimetric computation, calibration and maintenance is performed periodically using existing site procedures. Maintenance and test equipment, setting tolerances, calibration frequencies, and instrumentation accuracy were evaluated and accounted for within the thermal power uncertainty calculations.

*LEFM Inoperability*

In Attachment 3 to the application, the licensee provided a markup of the PBAPS Technical Requirements Manual (TRM), including a new Section 3.20, "Leading Edge Flow Meter (LEFM) System." With any of the three LEFM flow meters in the Maintenance mode but none in Fail mode, TRM 3.20 will allow operation at the TPO power level of ≤ 4,016 MWt for up to 72 hours. If all three LEFM flow meters are not in Normal mode at the end of 72 hours, power must be reduced to 4,010 MWt. If any of the LEFM flow meters are in the Fail mode, the flow input of the failed LEFM flow meter(s) to the core thermal power calculation must be replaced with the associated feedwater flow nozzle input. The LEFM flow meter(s) in Fail mode must then be restored to the Normal or Maintenance mode within 72 hours or power must be reduced to pre-MUR power level of ≤ 3,951 MWt.

The table below shows the proposed maximum allowable power levels for each LEFM mode after the 72-hour completion time is exceeded:

| LEFM Flow Meter Operating Mode | Total Power Uncertainty % | Maximum Power Level (MWt) |
|---|---|---|
| CheckPlus<br>(All in Normal) | 0.34% | 4,016 |
| Check<br>(One or More in Maintenance and None in Fail) | 0.5% | 4,010 |
| Fail<br>(Any in Fail) | 2.0% | 3,951 |

The licensee stated that the 72-hour allowed outage time (AOT) for the LEFM flow meters prior to reducing to the intermediate power level of 4,010 MWt or to the CLTP power level of 3,951 MWt are acceptable because the existing feedwater flow nozzle-based signals will be calibrated to the last validated data from the LEFM system during this period. Any slight drift of the feedwater flow nozzle measurements due to fouling would result in a higher than actual indication of feedwater flow and an overestimation of the calculated calorimetric power level. The licensee stated that this is conservative since the reactor will actually be operating below the calculated power level.

Based on the above discussion and the NRC staff's review of the licensee's submittals, the NRC staff concludes that the licensee provided sufficient justifications for the 72-hour AOT and the proposed power reduction actions if the AOT is exceeded.

*Criterion 1 Conclusion*

Based on the above considerations, the NRC staff concludes that the licensee has adequately addressed Criterion 1.

*Criterion 2*
*For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.*

The licensee stated that the PBAPS LEFM system installed instrumentation is bounded by the analysis and assumptions set forth in Topical Report ER-80P. The PBAPS LEFM system is currently installed and has been highly reliable. The maintenance history of the LEFM system since January 2011 identified three occasions where repairs took more than 72 hours to return an LEFM flow meter from the Maintenance mode to the Normal mode of operation.

Recommended maintenance practices from the LEFM system vendor that have changed since original installation of the LEFM system have been appropriately incorporated for implementation at PBAPS. Preventive maintenance scope and frequency is based on vendor recommendations and performance data reviews. Transducers are replaced as determined to be necessary by a review of the equipment's performance history by the LEFM system vendor. The operational and maintenance history of these components shows that the system is reliable for feedwater flow measurement and thermal power calculations.

Based on the above considerations, the NRC staff concludes that the licensee has adequately addressed Criterion 2.

*Criterion 3*
*The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and the LEFM for comparison.*

The licensee stated that PBAPS LEFM system uncertainty calculation methodology is based on PBAPS plant setpoint methodology. As discussed in SE Section 3.11.7, the PBAPS setpoint methodology is based on NRC-approved General Electric Topical Report NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996 (ADAMS Accession No. ML072950103, non-publicly available).

The method used to calculate LEFM system uncertainty is described in ASME PTC 19.1-1998, "Test Uncertainty, Instruments and Apparatus," and Caldon Engineering Report ER-80P (Reference 50), as supplemented by several other vendor reports.

Based on the above considerations, the NRC staff concludes that the licensee has adequately addressed Criterion 3.

<u>Criterion 4</u>
*For plant installation where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors are not representative of the plant-specific installation), licensees should provide additional justification for its use. The justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, licensees should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.*

The licensee stated that the LEFM system was installed at PBAPS after the receipt of the initial MUR license amendment in 2002. Although it has not been credited in the safety analyses since implementation of the EPU license amendment, the LEFM system remains the primary system to measure feedwater flow and provide input to the core thermal power calculator. The calibration factors for the PBAPS LEFM flow meter spool pieces were established by tests of these spools at Alden Research Laboratory in May 2002. These included tests of a full-scale model of the PBAPS hydraulic geometry and tests in a straight pipe. The licensee further stated that the piping configuration at PBAPS remains bounded by the original LEFM flow meter installation and calibration assumptions as analyzed in the Cameron topical reports.

The NRC staff notes there is a difference between the LEFM uncertainty level that was credited to support the original PBAPS MUR in 2002 and the currently proposed uncertainty values resulting from the calculations provided. To gain an understanding of the causes for change in LEFM uncertainty, the NRC staff sent an RAI to the licensee. In response to the RAI, the licensee in its supplement dated July 13, 2017, stated that for both submittals, an analytical uncertainty value was calculated with a corresponding allowable thermal power level. That power level was rounded down to the nearest whole megawatt, which effectively produced a slightly higher "credited" uncertainty. In 2002, the analyzed uncertainty was 0.36 percent, and the credited uncertainty was 0.38 percent. In the current submittal, the analyzed uncertainty is 0.34 percent, and the credited uncertainty is 0.36 percent to support a 4,016 MWt power level. The difference in uncertainties between the 2002 MUR and the current MUR LAR is 0.02 percent.

As discussed in the supplement dated July 13, 2017, the largest contributor to the 0.02 reduction in analyzed measurement uncertainty arises from a correction made to the application of a steam table enthalpy correlation uncertainty term.

Part of the 0.02 percent LEFM uncertainty difference is caused by a difference in the uncertainty associated with the volumetric flow measurement. The original MUR uncertainty calculation used a worst case guaranteed value from the manufacturer because operating data was not available at the time. The current submittal uses test data obtained from the PBAPS installed equipment and thus provides a slightly different uncertainty result even though the same equipment is being used.

The remaining difference in uncertainty is composed of a number of small contributors, such as feedwater temperature measurement. This difference is also attributable to use of available operational data for the PBAPS installed equipment for the current calculation in lieu of manufacturer supplied guaranteed uncertainties used in the original calculation. The NRC staff reviewed the LEFM uncertainty calculations and determined that the results have been developed consistent with the NRC-approved setpoint methodology.

Based on the above considerations, the NRC staff concludes that the licensee has adequately addressed Criterion 4.

*Criterion 5*
*Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.*

Based on the discussion in the "LEFM Inoperability" section under Criterion 1, the NRC staff concludes that the licensee has adequately addressed Criterion 5.

*Criterion 6*
*A CheckPlus operating with a single failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at an increased uncertainty.*

The licensee stated that when the LEFM CheckPlus meter on any of the three PBAPS LEFM flow meters has only one of its two LEFM Check subsystems fully operational resulting in that meter computing flow from just the remaining operational LEFM Check subsystem, that LEFM flow meter is considered in the Maintenance mode. This status is indicated to operators on the PMS computer in the control room. The total thermal power uncertainties for the three LEFM flow meter conditions in the new TRM Section 3.20 have been quantified on a plant-specific basis in accordance with the analysis in Cameron Engineering Reports ER-464 (Unit 2) and ER-463 (Unit 3) provided in the licensee's supplement dated March 20, 2017, and in accordance with the methodology in Caldon Engineering Report ER-157P (Reference 51). The licensee provided the information in the following table to show the results of these analyses with the highest uncertainty for each condition.

| LEFM Flow Meter Condition | Total Power Uncertainty % | Associated Power Level (MWt) |
|---|---|---|
| All in Normal mode | 0.34% | 4,016 |
| One or more in Maintenance mode, none in Fail mode | 0.5% | 4,010 |

Based on the above considerations, the NRC staff concludes that the licensee has adequately addressed Criterion 6.

*Criterion 7*
*An applicant with a comparable geometry can reference the findings in Section 3.2.1 of the NRC staff's SE Caldon Engineering Report ER-157P (Reference 53) to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.*

The NRC determined that for conditions in which the CheckPlus system is operating with one or more transducers out of service, the effect of downstream piping should be addressed if the separation distance from the meter transducers to the downstream piping change is less than five pipe diameters. The licensee stated that, at PBAPS, the LEFM flow meters are installed upstream of the feedwater flow nozzles, and the distance from meter transducers to downstream piping changes (i.e., venturi contraction), is greater than five pipe diameters in each feedwater line. Therefore, the downstream geometries for PBAPS do not have a significant influence on CheckPlus calibration.

Based on the above considerations, the NRC staff concludes that the licensee has adequately addressed Criterion 7.

*Criterion 8*
*Any applicant that requests an MUR with the upstream flow straightener configuration discussed in Section 3.2.2 of the NRC SE for Caldon Engineering Report ER-157P (Reference 53) should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Caldon Engineering Report ER-790 (ADAMS Accession No. ML100840025). Since the evaluation in Caldon Engineering Report ER-790 does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.*

The installed configuration of the PBAPS LEFM flow meters does not include an upstream flow straightener. Therefore, this criterion is not applicable to the PBAPS MUR application.

*Criterion 9*
*An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of Caldon Report ER-764 (ADAMS Accession No. ML100820167).*

The licensee does not assume large uncertainties in steam moisture content. The PBAPS moisture separators and dryers are expected to deliver steam with low moisture content, and the uncertainty calculation assumes a steam quality as a constant moisture fraction of 0.0 percent. Therefore, no engineering basis for the distribution of the uncertainties in steam moisture is required.

Based on the above considerations, the NRC staff concludes that the licensee has adequately addressed Criterion 9.

Based on the evaluation discussed above, the NRC staff concludes that the licensee has sufficiently addressed the NRC criteria for use of Caldon Topical Reports ER-80P and ER-157P.

*Plant-Specific Power Uncertainty Analysis*

As discussed in Section 4.1 of the NRC staff's SE for the TLTR, the generic evaluations and discussions in the TLTR are based on a power uprate of less than or equal to 1.5 percent. However, plant-specific applications could request a higher TPO uprate (e.g., 1.7%), depending on the plant-specific feedwater flow measurement uncertainty. TPO uprate submittals that are based on a power level uncertainty of less than 0.5 percent need to be addressed on a plant-specific basis.

As discussed in Section 3.2 of Attachment 1 to the licensee's application, with credit for the LEFM system, the core thermal power measurement uncertainty will be a maximum of 0.34 percent. As such, the licensee stated this will support an increase in RTP of 1.66 percent (i.e., 2.00 – 0.34) from 3,951 MWt to 4,016.6 MWt, which is conservatively rounded down to the requested value of 4,016 MWt. Since the proposed power uprate is greater than 1.5 percent, the licensee has provided the following plant-specific analyses to justify the power measurement uncertainty:

- Cameron Engineering Report ER-464, Revision 5, "Uncertainty Analysis for Thermal Power Determination at Peach Bottom Unit 2 Using the LEFM✓+ System," dated February 2017.

- Cameron Engineering Report ER-463, Revision 5, "Uncertainty Analysis for Thermal Power Determination at Peach Bottom Unit 3 Using the LEFM✓+ System," dated February 2017.

Proprietary versions of the above-listed Cameron reports were provided in Attachment 1 to the licensee's supplement dated March 20, 2017 (Reference 2). Non-proprietary versions of the reports were provided in Attachment 2 to the licensee's supplement dated March 20, 2017.

The NRC staff reviewed the Cameron reports and determined that all the parameters associated with the thermal power measurement uncertainty, individual measurement uncertainties, and calculated the overall thermal power uncertainty were properly identified.

The PBAPS setpoint methodology is based on NRC-approved General Electric Topical Report NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996 (ADAMS Accession No. ML072950103, non-publicly available). This methodology statistically combines individual uncertainties to determine the overall uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are arithmetically combined to form statistically independent groups, which are then added to the square root of the sum of the squares for the remaining random parameters to determine the overall uncertainty. The vendor's determination of the uncertainty of the Cameron LEFM CheckPlus system is consistent with this methodology.

Based on the above considerations, the NRC staff concludes that the licensee has adequately justified the power measurement uncertainty for the proposed TPO uprate.

## 3.6 TSAR Section 6.0 – Electrical Power and Auxiliary Systems

The following provides the NRC staff's technical review of the topics in Section 6.0 of the TSAR.

### 3.6.1 TSAR Section 6.1 – AC [Alternating Current] Power

*TSAR Section 6.1.1 – Offsite Power (Power Block Equipment)*

This section of the NRC staff's SE addresses the power block equipment (e.g., generator, transformers, switchyard components, and isolated phase bus duct) as discussed in TSAR Section 6.1.1.

As discussed in Section 8.2 of the UFSAR, the PBAPS main generator manufactured by General Electric Company is a three-phase, 60 Hertz (Hz), 22 kilovolt (kV) generator operating at 1,800 revolutions per minute (rpm) with 75 pounds per square inch gage (psig) of hydrogen pressure. As discussed in TSAR Section 6.1.1, the PBAPS, Unit 2, generator is rated at 1,530 megavolt amp (MVA) with a 0.92 power factor and a 0.510 short circuit ratio. The TSAR also states that the Unit 3 generator is rated at 1,530 MVA with a 0.90 power factor and a 0.540 short circuit ratio.

In an RAI, the NRC staff noted that the projected electrical power output for the Unit 3 main generator (1,387.9 megawatts electric (MWe), as shown in TSAR Section 7.1) exceeds the generator rating (1,377 MWe, as shown in TSAR Table 6-2). The staff requested the licensee to explain how the main generator for Unit 3 will operate within its generating capability curve if its electrical power output is higher than its electrical rating. In the supplement dated July 13, 2017, the licensee stated that the main generators will not exceed their design ratings due to the fact that the projected maximum TPO operating point, at the lowest historical power factor since January 2014, is 1,388 MWe at 0.945 power factor. The licensee stated that at this power factor of 0.945, and the maximum TPO winter output at 1,388 MWe, the generator would operate at 1,469 MVA. As noted above, both Unit 2 and Unit 3 generators are rated at 1,530 MVA. Based on review of the supplemental information, the NRC staff concludes that the Unit 3 main generator would operate within the generator capability curve based on the fact that the maximum projected operating point of the main generator (i.e., 1,469 MVA) is less than the rating of the generator (i.e., 1,530 MVA).

Additionally, in TSAR Section 6.1.1 the licensee stated, "For summer and winter operations, the gross generator MWe output will be kept on, or within, the existing generator reactive capability curve." In the supplement dated July 13, 2017, the licensee stated that the projected gross generator output for both Units 2 and 3 at licensed TPO power level is approximately 1,388 MWe/1,469 MVA (winter operation), and 1,323 MWe/1,400 MVA (summer operation), which is within the generator capability curves for both Units 2 and 3. In the supplement, the licensee stated that the electrical output and impact on generator protective devices is currently being evaluated for TPO. The licensee stated that the generator protective relay settings will be changed in accordance with Exelon's configuration control program to support the increased generator MWe output with the TPO uprate if necessary. Based on the above considerations, the NRC staff concludes that there is reasonable assurance that the main generators are adequate to support the TPO uprate.

In TSAR Section 6.1.1, the licensee stated that the PBAPS 1,530 MVA main power transformer consists of three single-phase, oil-directed, air-forced, 60 Hz, oil-filled type, outdoor transformers. The TSAR also states that the main transformers and the associated switchyard components (rated for maximum generator output) are adequate for the TPO uprate-related transformer output and remain bounded with the TPO review. The licensee stated that the items with the least margin are the Unit 2 disconnect switches, which have 25.6 percent margin. Based on the main transformer's operating output remaining within its nameplate rating and the associated switchyard components remaining unchanged, the NRC staff concludes that the main power transformer system and the associated switchyard components are adequate to support the TPO uprate.

In TSAR Section 6.1.1, the licensee stated that the isolated phase bus duct consists of a main bus, a delta bus, and an auxiliary bus. In addition, the TSAR states that both rated voltage and low voltage current output is adequate for the isolated phase bus duct. In the supplement dated July 13, 2017, the licensee provided the ratings of the isolated phase bus duct sections and the load values at the maximum rated generator output of 1,530 MVA. The licensee reviewed the ratings and load values for each section of the isolated phase bus duct and determined that load values at maximum rated generator output bound the projected load values at TPO uprate conditions. The NRC staff reviewed the load values on the generator bus, main section, delta section, and auxiliary sections to determine if the isolate phase bus duct is adequately rated for TPO uprate conditions. Based on the ratings provided in the supplement, the NRC staff concludes that the isolated phase bus duct system capacity is adequate for the expected loads during the TPO uprate operation because the load values at 1,530 MVA remain within the ratings of the individual sections of the isolate phase bus duct. Therefore, the NRC staff further concludes that the isolated phase bus duct system is adequate to support the TPO uprate.

Based on the above discussion, the NRC staff concludes that the main generator, main transformer, associated switchyard components, and isolated phase bus remain adequately sized and have adequate capacity for TPO conditions. The staff further concludes that the licensee has adequately addressed the impact of the TPO uprate on the power block equipment.

*TSAR Section 6.1.1 – Offsite Power (Grid Studies)*

This section of the NRC staff's SE addresses the effect of the proposed TPO uprate with respect to electrical grid stability as discussed in TSAR Section 6.1.1.

In TSAR Section 6.1.1, the licensee stated that there is no significant effect on grid stability or reliability because there were no modifications associated with the TPO uprate that would increase electrical loads beyond those levels previously included or that would require revising the logic of the distribution systems. A grid stability study and a voltage analysis for the TPO uprate was provided in Attachment 13 to the application. The licensee stated in TSAR Section 6.1.1 that the grid stability study considered the increase in electrical output to demonstrate conformance to final GDC 17.

In TSAR Section 6.1.1, the licensee stated that the grid voltage analysis, which was performed by the Philadelphia Electric Company (PECO), verified the transmission system's capability to maintain the post-trip voltage drops and voltages at the safety buses above the reset value of the degraded voltage relay (DVR) on a steady-state basis. The licensee provided additional information regarding the grid voltage analysis in the supplement dated July 13, 2017. This information included the DVR reset voltage value, the post-MUR PECO maximum voltage drop results, and the maximum voltage drops from PBAPS procedure SE-16, "Grid Emergency," which provides the maximum post-trip contingency percentage voltage drop for each offsite power source.

The licensee stated in the supplement dated July 13, 2017, that as long as the post-trip contingency percentage voltage drops are less than the limits under accident conditions, the DVRs do not time out and cause the safety-related buses to be separated from the offsite sources. In addition, the licensee stated that under the worst case offsite source voltage drops, as predicted in the PECO grid voltage analysis, the PBAPS safety buses will remain connected to the offsite source(s). The NRC staff reviewed the supplemental information and compared the post-MUR PECO maximum voltage drop results to the maximum voltage drops from procedure SE-16 and concluded there is reasonable assurance that the transmission system will maintain the post-trip voltage drops and voltages at the safety buses above the reset value of the DVR at TPO operating conditions.

The NRC staff reviewed the grid stability study and the licensee's supplemental information, and concludes that there is reasonable assurance that the operation under the proposed TPO conditions will continue to provide stable and reliable grid operation due to the fact that there are no modifications associated with the TPO uprate that would increase electrical loads as stated in Section 6.1.1 of the TSAR.

*TSAR Section 6.1.2 – Onsite Power*

The alternating current (AC) distribution system is the source of power for the non-safety-related buses and for the safety-related emergency buses. According to the UFSAR Section 8.0, the normal power sources during plant operation for all auxiliary system buses except the emergency buses and cooling tower equipment are the main generators. Power is supplied from each generator's main 22 kV isolated-phase bus through a unit auxiliary transformer to the 13.8 kV unit auxiliary switchgear buses. The AC subsystems consist of the 13.8 kV and 13.2 kV normal electrical distribution system and 4.16 kV normal and essential auxiliary systems.

Table 8.4.1 in the UFSAR includes the reactor recirculation pumps and condensate pumps as loads that are on the 13.2 kV buses. According to Section 6.1.2 of the TSAR, the 13.2 kV load demands that will be affected are the reactor recirculation pumps in which the horsepower requirements of these pumps will increase approximately 0.08 percent due to the power uprate. Additionally, the licensee has stated that even though there are negligible changes in the onsite loads, operation at the TPO RTP level is achieved in both normal and emergency conditions by operating equipment at or below the nameplate rating brake horsepower.

Based on the above, the NRC staff concludes that the load changes to the onsite AC power distribution system design-basis loads, voltage regulation, or reduction in the design margins due to the TPO uprate conditions are minimal and will not adversely impact the loadings and voltages of the safety-related buses. Therefore, the NRC staff concludes that the AC power system has adequate capacity to operate the plant equipment within its design to support the TPO power uprate.

*TSAR Section 6.1.3 – Emergency Diesel Generator*

According to UFSAR Section 8.5.3, the standby AC power source consists of a total of four emergency diesel generators (EDGs) serving both PBAPS, Units 2 and 3. Each EDG consists of a diesel engine, a generator, and the associated auxiliaries mounted on a common base. The EDG system automatically supplies emergency electrical power to the engineered safeguard features plus selected non-safety loads in the event that the normal AC power is interrupted.

The licensee stated in TSAR Section 6.1.3 that there are no load increases to the emergency buses loads supported by the EDGs due to the TPO uprate. The NRC staff concludes that the EDGs will continue to have adequate capacity and capability to power the safety-related loads at TPO power uprate conditions due to that fact that there are no load increases on the safety busses.

### 3.6.2 TSAR Section 6.2 – DC [Direct Current] Power

As discussed in UFSAR Section 8.7, the direct current (DC) power system is composed of station batteries, battery chargers, and the DC distribution system. It provides DC power to motive, control, instrumentation, and other DC loads.

The licensee stated in TSAR Section 6.2 that there are no changes to the DC power system loading resulting from the TPO. Therefore, the NRC staff concludes that the DC power system has adequate capacity to operate the plant equipment within its design to support the TPO power uprate.

### 3.6.3 TSAR Section 6.3 – Fuel Pool

The description of the spent fuel pool (SFP) cooling system provided in Section 6.3 of the TSAR indicated that the licensee maintains the heat load within the capability of the fuel pool cooling and cleanup system (FPCCS) through cycle-specific calculations. The calculations verify that the SFP heat load would allow the SFP bulk temperature to be maintained less than or equal to the currently analyzed temperature limit of 150 °F as stated in UFSAR Section 10.5.3.

In the supplement dated July 13, 2017, the licensee clarified that the operators use procedures to manage the decay heat rate by controlling the timing of spent fuel transfer to the SFP to ensure the maximum SFP temperature remains within the 150 °F licensing basis limit. The TPO uprate does not affect the heat removal capability of the FPCCS used for normal refueling or the FPCCS supplemented with the residual heat removal system (RHR) fuel pool cooling assist mode for full core discharges. The TSAR stated that the TPO heat load would remain within the design-basis heat load for the FPCCS supplemented with RHR assist mode. Table 6-7, "FPCCS Response at CLTP and CLTP x 1.02," of the TSAR showed the peak SFP temperature could be managed to stay within SFP temperature limits for scenarios where the cooling water temperature was high and the decay heat rate was based on operation at 102 percent of CLTP, which bounds the decay heat that would result from TPO uprate power. Thus, the NRC staff concludes that the decay heat load resulting from operation at TPO uprate conditions is acceptable and can be maintained within the capacity of the FPCCS for normal discharges and the FPCCS supplemented by the RHR SFP cooling assist mode for full core discharges.

3.6.4 TSAR Section 6.4 – Water Systems

The PBAPS cooling water systems include service water (SW), emergency service water (ESW), high-pressure service water (HPSW), turbine building closed cooling water (TBCCW), reactor building closed cooling water (RBCCW), and chilled water (CW). Of these systems, the ESW and HPSW systems are classified as safety-related. The performance of the safety-related SW systems during and following design-basis events does not change because the current analyses were performed at 102 percent of CLTP.

The normal heat sink for PBAPS is the Conowingo Pond. PBAPS also has an emergency heat sink facility that provides cooling water to the ESW and HPSW systems when the Conowingo Pond is not available. The emergency heat sink consists of a multi-cell induced draft cooling tower over a large water storage basin with an adequate inventory to achieve and maintain safe shutdown of both units for 7 days without makeup. The analysis presented in Section 6.4 of the TSAR indicates that the basin margin is reduced from about 17.3 percent at CLTP to about 15.7 percent at TPO conditions. Therefore, the emergency heat sink facility remains adequate for TPO operation.

The non-safety-related SW system would experience an increase in discharge temperature due to power-related increases in heat rejection to the TBCCW, RBCCW, and CW systems. The major SW heat load increase from the TPO reflects an increase in main generator losses rejected to the stator water and hydrogen coolers to the TBCCW system. The TSAR noted that the RBCCW system may experience a small increase as a backup heat removal path from the FPCCS, but the change in heat removal would be negligible. The CW system consists of the non-safety-related drywell CW and control room CW systems. The heat load change resulting from TPO is negligible for these systems. Therefore, the plant-specific analyses described in Section 6.4 of the TSAR indicates that the increase in heat loads resulting from TPO operation is within the available margin for the SW, TBCCW, RBCCW, and CW systems.

Operation at TPO uprate conditions increases the heat rejected to the condenser, which could affect the condenser vacuum. The plant-specific evaluation of the main condenser at TPO conditions confirmed that the condenser and the associated circulating water system are adequate for TPO operation.

The NRC concludes that based on the licensee's evaluation provided in TSAR Section 6.4, there is reasonable assurance that the TPO uprate will not impact the ability of the water systems from performing their intended functions.

3.6.5 TSAR Section 6.5 – Standby Liquid Control System

The SLC system provides the alternate means of attaining and maintaining cold shutdown conditions, assuming no control rod movement. The system is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that all or some of the control rods cannot be inserted. This manually operated system pumps a highly enriched sodium pentaborate solution into the vessel to achieve a subcritical condition.

As discussed in TSAR Section 6.5, a plant-specific evaluation concluded that the TPO uprate does not affect shutdown or injection capability of the SLC system. Because the shutdown margin is reload dependent, the shutdown margin and the required reactor boron concentration are confirmed for each reload core. The plant-specific ATWS evaluation shows that the maximum reactor lower plenum pressure following the limiting ATWS event reaches 1,207 pounds per square inch atmospheric (psia) during the time the SLC system is analyzed to be in operation. This evaluation shows the pressure margin for the SLC pump discharge relief valves is 183 psi, which is adequate to ensure that the SLC system relief valves remain closed during system injection. Therefore, the licensee concluded that SLC system operation during an ATWS at the TPO power level is acceptable, considering the MELLLA+ operating domain expansion, and it will continue to meet the requirements of 10 CFR 50.62.

The NRC concludes that based on the licensee's evaluation provided in TSAR Section 6.5, there is reasonable assurance that the TPO uprate will not impact the ability of the SLC system from performing its intended function.

3.6.6 TSAR Section 6.6 – Power-Dependent Heating, Ventilation, and Air Conditioning

Section 6.6 of the TSAR presented an evaluation of heating, ventilation, and air conditioning (HVAC) systems that are potentially affected by the TPO uprate, which consist of heating, cooling supply, exhaust, and recirculation units in the turbine building, reactor building, steam tunnel, and primary containment (drywell). The evaluation indicated that the TPO uprate would result in a minor increase in the MS tunnel heat load caused by the slightly higher feedwater temperature. Other areas experience a negligible increase or no change in heat load.

The NRC concludes that based on the licensee's evaluation provided in TSAR Section 6.6, there is reasonable assurance that the TPO uprate will not impact the ability of the power-dependent HVAC systems from performing their intended functions.

3.6.7 TSAR Section 6.7 – Fire Protection

The NRC staff's review with respect to fire protection focused on the effects of the LAR on the post-fire safe shutdown capability and increase in decay heat generation following plant trips.

As discussed in TSAR Section 6.7, "Fire Protection," the licensee stated that the operation of the plant at the TPO RTP level does not affect the fire suppression or detection systems. There is no change in the physical plant configuration or combustible loading resulting from the TPO uprate. Further, the licensee stated that the operator manual actions that are used for compliance with 10 CFR Part 50, Appendix R, were reviewed. No new operator actions have been identified in areas where environmental conditions such as heat would challenge the operator or would become a challenge with TPO conditions. Because this uprate is being performed at a constant pressure and temperature, the normal temperature environments are not affected by TPO. Therefore, the operator manual actions required to mitigate the consequences of a fire are not affected.

The licensee conducted a review of the fire protection program related to administrative controls, fire barriers, fire protection responsibilities of plant personnel, and resources necessary for systems required to achieve and maintain safe shutdown. The review focused on the effect of TPO uprate and how it would affect these areas. The licensee concluded that the TPO uprate will have no effect on fire protection administrative controls, fire barriers, fire protection responsibilities of plant personnel and resources necessary for systems required to achieve and maintain safe shutdown.

The NRC staff has reviewed the licensee's statements in the LAR related to the impact of the MUR power uprate on plant safe shutdown and impacts due to increase in decay heat. For the MUR power uprate, the licensee reviewed its systems to achieve and maintain the plant in cold shutdown condition. The licensee stated that the current PBAPS Appendix R analysis demonstrates that the plant can reach cold shutdown with significant margin to the 72-hour requirements stated in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L. The licensee concluded that the additional decay heat removal as a result of the TPO uprate would not affect the ability to reach and maintain cold shutdown within 72 hours.

The NRC staff reviewed the LAR information concerning use of fire protection water for non-fire suppression activities. The licensee stated that the PBAPS does not take credit in any safety analysis for the fire protection system other than for fire protection activities. Further, the licensee stated that procedures are provided under the Transient Response Implementation Plan, Severe Accident Management Guidelines, Extensive Damage Mitigation Guidelines, and FLEX Support Guidelines, which provide instructions for utilizing fire protection system pumps to provide water to the reactor, the drywell, the spent fuel pool, or the suppression chamber if necessary. However, this use of the non-safety-related fire protection system is not credited in any safety analyses and TPO uprate operation will not require any changes to these procedures regarding the utilization of the fire protection system.

As discussed in TSAR Section 6.7.1, "10 CFR 50 Appendix R Fire Event," the licensee stated that a plant-specific analysis for PBAPS at TPO RTP conditions was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR Part 50 Appendix R. The licensee reanalyzed two limiting shutdown methods, A and C, under TPO conditions[5]. The bounding peak cladding temperature, for PBAPS was reevaluated for shutdown Method C with one residual heat removal (RHR) train in low-pressure coolant injection (LPCI) mode. The bounding peak suppression pool temperature for PBAPS was reevaluated for shutdown Method A with reactor core isolation cooling, and one RHR train in LPCI mode.

The licensee stated that the results of the Appendix R evaluation at TPO conditions (as shown in TSAR Table 6-8) demonstrate that the fuel cladding integrity, reactor vessel integrity, and containment integrity are maintained. One train of systems remains available to achieve and maintain safe shutdown conditions. For fire safe shutdown (FSSD) Method C1, the time from event initiation to the point at which conditions require manual initiation of RPV depressurization from the control room is slightly reduced, but is more than the time available for other FSSD methods requiring manual RPV depressurization. The time line for manual operator actions, including FSSD Method C1, and the associated approved 10 CFR Part 50 Appendix R exemptions required for FSSD analysis are unaffected. There are no required changes to the operating procedures or implementing strategy for the TPO uprate.

The NRC staff finds that the information provided in the LAR, as described above, demonstrates that compliance with the fire protection and safe shutdown program will not be affected because the MUR power uprate evaluation did not identify changes to design or operating conditions that will adversely impact the post-fire safe shutdown capability. MUR uprate does not change the credited equipment necessary for post-fire safe shutdown, nor does it require reroute of essential cables, relocation of essential components/equipment, or introduction of new or

[5] The shutdown methods are discussed in Section 5.2 of the PBAPS Fire Protection Plan, which is included as part of the UFSAR.

changes to existing operator manual actions credited for post-fire safe shutdown. The licensee has made no changes to the plant configuration or combustible loading as a result of modifications necessary to implement the MUR power uprate that affect the PBAPS fire protection program. The NRC staff agrees with the licensee's conclusion that the proposed power uprate will not have adverse effects on post-fire safe shutdown capability of the plant. Therefore, the NRC staff concludes that the LAR is acceptable with respect to fire protection.

#### 3.6.8 TSAR Section 6.8 – Systems Not Affected by TPO Uprate

Appendix A to the TLTR presents the proposed format of a plant-specific TSAR. The TSAR for PBAPS is consistent with the format and content of the systems to be evaluated for a TPO uprate. As such, the NRC staff concludes that all systems that are significantly affected by TPO have been addressed in the TSAR. Based on previous NRC reviews, the staff further concludes that systems not addressed by the TSAR are not significantly affected by the TPO uprate.

### 3.7 TSAR Section 7.0 – Power Conversion Systems

The licensee provided an evaluation of the TPO uprate effects on non-safety-related power conversion systems in Section 7.0 of the TSAR, including the turbine-generator, turbine steam bypass system, and the FW and condensate system. Section 3.5.2 of this SE addresses control systems associated with these mechanical systems. The turbine-generator is important to safety with respect to turbine missile generation probability, but that analysis is not affected by the TPO uprate because the turbine speed remains constant and the turbine operating conditions do not significantly change. Turbine bypass capability is considered in the analyses for generator load rejection and turbine trip transients and is unchanged by the TPO uprate. Furthermore, the limiting transient condition assumes failure of turbine bypass. Also, the effect of the TPO uprate on turbine bypass capacity measured as a fraction of total steam flow is insignificant, so the turbine bypass capacity remains acceptable. The FW and condensate systems have adequate margin to accommodate the minor flow increase associated with the TPO uprate.

The NRC concludes that based on the licensee's evaluation provided in TSAR Section 7.0, there is reasonable assurance that the TPO uprate will not impact the ability of the power conversion systems from performing their intended functions.

### 3.8 TSAR Section 8.0 – Radwaste and Radiation Sources

The following provides the NRC staff's technical review of the topics in Section 8.0 of the TSAR.

#### 3.8.1 TSAR Section 8.1 – Liquid and Solid Waste Management

Section 8.1 of the TSAR addressed the liquid and solid radioactive waste systems, which collect, process, and store radioactive waste for reuse, discharge of liquids, and shipment. The primary source of liquid and solid radioactive waste is from the condensate filter/demineralizers (CFD). The TPO uprate results in approximately a 2 percent increase in condensate flow causing a possible reduction in average time between back washes of CFD resin. The activated corrosion products in the waste stream will increase in proportion to TPO uprate caused by increase in power and flow through the CFD. The other systems contributing to liquid and solid waste experience no significant changes, and the total volume of processes waste is not expected to increase appreciably as a result of operation at TPO conditions. These factors have no affect plant safety and have minimal effect on the radioactive waste system.

The NRC concludes that based on the licensee's evaluation provided in TSAR Section 8.1, the liquid and solid radioactive waste handling capabilities are satisfactory for TPO.

3.8.2 <u>TSAR Section 8.2 – Gaseous Waste Management</u>

Section 8.1 of the TSAR addressed the gaseous radioactive waste management systems, which include the offgas system and the various building ventilation systems, which collect, control, and process gaseous radioactive waste. The TPO increase in power will cause more radiolytic decomposition of water into hydrogen and oxygen, causing a higher heat load on offgas components. However, the increase is well within the design of the offgas system, and, thus, does not cause any adverse effect on the offgas system. Therefore the TPO uprate does not significantly affect the offgas system design or operation.

The NRC concludes that based on the licensee's evaluation provided in TSAR Section 8.2, the gaseous radioactive waste handling capabilities are satisfactory for TPO.

3.8.3 <u>TSAR Sections 8.3 through 8.6 – Radiological Consequences for Normal Operation</u>

As discussed in Section 1.2.3 of the TSAR, the licensee evaluated the proposed MUR with respect to radiological consequences for normal operation including the effect of source terms, onsite doses and offsite doses. The licensee's evaluations are described in TSAR Sections 8.3 through 8.6. TLTR Appendix H describes the methodologies and assumptions for the licensee's evaluation of the radiological effects of the proposed MUR.

The NRC staff conducted its review in this area to ascertain what overall effects the proposed MUR will have on both normal occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable (ALARA). The NRC's acceptance criteria for normal occupational and public doses are based on: (1) 10 CFR Part 20, 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; and (2) 10 CFR Part 50, Appendix I, which establishes numerical guides for design objectives and limiting conditions for operation to meet the ALARA criterion.

*Radiation Sources in Reactor Core and Reactor Coolant*

As discussed in Section 5.4 of the TLTR, during normal operation, the radiation levels in the plant are the result of direct and scattered radiations from the reactor core and from radioisotopes carried in the reactor water, steam, or radwaste process.

As discussed in TSAR Section 8.3, 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 reactions as a secondary result of fission. For TPO, the percent increase in the operating source terms is no greater than the percent increase in power. The licensee further stated that the PBAPS-specific source term increases due to the TPO uprate are bounded by the safety margins of the design-basis sources.

The reactor coolant isotopic activity concentration is a function of the core power level, the migration of radionuclides from the fuel, radioactive decay, and the removal of radioactive material by coolant purification systems. As discussed in TSAR Section 8.4, radiation sources in the reactor coolant include activation products, activated corrosion products, and fission products.

With respect to the activation products, the licensee stated that because the sources are produced by interactions in the core region, their rates of production are proportional to power. However, the concentration in the steam remains nearly constant because the increase in activation production is balanced by the increase in steam flow.

With respect to activated corrosion products, the licensee stated that under the TPO uprate conditions, the activation rate in the reactor region increases with power and the filter efficiency of the condensate demineralizers may decrease. The net result may be an increase in the activated corrosion product production. However, total TPO activated corrosion product activity levels in the reactor water remain less than the design-basis activated corrosion product activity.

With respect to fission products, the licensee stated that activity levels in the reactor water at TPO conditions are approximately equal to current measured data, which are fractions of the design-basis values.

Based on the information in TSAR Section 8.3 and 8.4 discussed above, the NRC staff concludes that the source terms under TPO will remain bounded by the current design basis.

*Radiation Levels and Normal Operation Onsite and Offsite Doses*

As discussed in TSAR Section 8.5, the licensee reviewed the projected radiation exposure at normal operation radiation levels for the TPO uprate. Normal operation radiation levels increase slightly for the uprate. However, the licensee stated that the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the design, source terms, and analytical techniques. In addition, the licensee stated that the PBAPS site ALARA program will maintain individual worker exposures within acceptable limits.

There are several factors associated with a power uprate that may impact public and offsite radiation exposures during plant operations. These include a possible increase in gaseous and liquid effluents released from the site, and the possible increase in offsite radiation exposure from radioactive plant components onsite, either directly or from atmospheric scatter (known as skyshine). As discussed in TSAR Section 8.6, the licensee stated that at CLTP, the public dose effects of design-basis gaseous and liquid releases remain within 10 CFR Part 50, Appendix I limits with substantial margin, In addition, the uprate does not involve significant increases in the offsite dose from noble gases, airborne particulates, iodine, tritium, or liquid effluent. Furthermore, radiation from skyshine is not a significant exposure pathway. The licensee also stated that present offsite radiation levels are a negligible portion of background radiation.

*Radiological Consequences for Normal Operation Conclusion*

The NRC staff has reviewed the licensee's assessment of the effects of the proposed MUR with respect to radiological consequences for normal operation. Based on the information in TSAR Sections 8.3 through 8.6, as discussed above, the staff concludes that there is reasonable assurance that occupational and public radiation doses will not be significantly affected by normal operation at the MUR power level and will continue to remain below applicable limits in 10 CFR Part 20 and Appendix I to 10 CFR Part 50.

3.9 <u>TSAR Section 9.0 – Reactor Safety Performance Evaluations</u>

The following provides the NRC staff's technical review of the topics in Section 9.0 of the TSAR.

3.9.1 <u>TSAR Section 9.1 – Anticipated Operational Occurrences</u>

Anticipated operational occurrences (AOOs) are abnormal transients that are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error.

Chapter 14 of the PBAPS UFSAR contains the design-basis analyses that evaluate the effects of an AOO resulting from changes in the system parameters such as: (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The facility's responses to the most limiting transients are analyzed each reload cycle, and corresponding changes in the MCPR are added to the SLMCPR to establish the operating limit MCPR. These thermal limits are determined by the licensee based on an NRC-approved methodology as discussed in SE Section 3.2.2.

As discussed in TSAR Section 9.1, **[[**

**]]**. The standard reload analyses consider the plant conditions for each fuel cycle.

Because the licensee will perform the reload analysis at the TPO uprated conditions using an NRC-approved methodology and the thermal limits will be maintained for operation at the uprated conditions during AOOs and accidents, to ensure the fuel cladding integrity, the NRC staff finds the licensee's evaluation with respect to AOOs acceptable.

3.9.2 <u>TSAR Section 9.2 – Design Basis Accidents</u>

As discussed in Section 5.4 of the NRC staff's SE for the TLTR, "Radiological Consequences," the NRC staff found that there is reasonable assurance that there would be no increase in the radiological consequences due to postulated DBA events following a TPO uprate if the licensee's previous analyses were based on 102 percent of the CLTP and if all other analysis assumptions, inputs, and methodologies are unchanged.

As discussed in TSAR Section 9.2, the licensee stated that postulated DBA events have either been previously analyzed at 102 percent of the CLTP power level of 3,951 MWt (i.e., at 4,030 MWt), which bounds the TPO power level (i.e., 4,016 MWt), or the postulated events are not dependent on core thermal power. The licensee also stated that the evaluation/analysis was based on the methodology, assumptions, and analytical techniques described in the PBAPS current licensing basis, regulatory guides, and in previous SEs.

The NRC staff reviewed the current licensing basis DBA radiological consequence analyses, as documented in Chapter 14 of the UFSAR. The specific DBA analyses that were reviewed are as follows:

- UFSAR Section 14.9.2.1 – Loss-of-Coolant Accident;
- UFSAR Section 14.9.2.2 – Refueling Accident;
- UFSAR Section 14.9.2.3 – Main Steam Line Break Accident; and,
- UFSAR Section 14.9.2.4 – Control Rod Drop Accident.

The licensee's analyses shown in the UFSAR determined the doses for each of the DBAs at the exclusion area boundary (EAB), the outer boundary of the low population zone (LPZ); and in the control room. The results of the analyses for each of the DBAs is shown in UFSAR Table 14.9.6 (control room) and UFSAR Table 14.9.7 (EAB and LPZ). The licensee compared the results against the applicable dose acceptance criteria in 10 CFR 50.67 and guidance in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" (Reference 13). As discussed in UFSAR Section 14.9.2.5, the EAB, LPZ, and control room doses following the postulated DBAs are within the allowable regulatory dose limits.

The NRC staff also reviewed the LAR to confirm that the licensee has accounted for the potential for an increase in measurement uncertainty should the LEFM system experience operational limitations. As discussed in Section 3.3.4 of Attachment 1 to the LAR, the LEFM system features automatic self-testing. A continuously operating on-line test is provided to verify that the digital circuits are operating correctly and within the specified accuracy range. If the LEFM system or a portion of the system becomes inoperable, control room operators are promptly alerted by a Plant Monitoring System alarm in the control room. As shown in Attachment 3 to the LAR, the licensee has proposed to add new Section 3.20, "Leading Edge Flow Meter (LEFM) System," to the Technical Requirements Manual (TRM). This new TRM section is intended to ensure that core thermal power is maintained at a level consistent with the feedwater flow measurement uncertainty.

Based on the information described above, the NRC staff concludes that there is reasonable assurance that the current licensing basis dose consequence analyses performed at 102 percent of the CLTP level will remain bounding at the proposed MUR uprated power level of 4,016 MWt, with a margin that is within the assumed uncertainty associated with the LEFM system.

3.9.3 <u>TSAR Section 9.3 – Special Events</u>

*TSAR Section 9.3.1 – Anticipated Transients without Scram*

ATWS is an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR 50.62. For BWRs, 10 CFR 50.62 requires, in part, that:

- 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.

- Each BWR have an 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.

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

PBAPS meets the ATWS mitigation requirements defined in 10 CFR 50.62. PBAPS has an SLC system capable of boron injection equivalent to 86 gpm of a 13 weight percent natural boron equivalency and has installed an ARI system and automatic recirculation pump trip (RPT) logic.

BWR facilities are also analyzed against certain ATWS acceptance criteria to demonstrate their ability to withstand an ATWS event. These criteria include maintaining fuel integrity (the core and fuel must maintain a coolable geometry), primary system integrity (the peak reactor vessel pressure remains below 1,500 psig), and containment integrity (the containment temperature and pressure must not exceed the design limit).

Because plant-specific ATWS analysis for EPU and MELLLA+ were previously performed at the CLTP, an additional plant-specific analysis of the limiting ATWS events was performed at the TPO bounding high thermal power of 4,018 MWt (101.7 percent). The TPO RTP ATWS analysis was performed using the NRC-approved computer codes PANAC, ODYN, TASC, and TRACG.

M+ LTR Limitation and Condition 12.23.2 requires that the plant-specific automatic settings be modeled for ATWS. For PBAPS, the plant automatic settings, which include the ATWS-RPT, low-pressure isolation, and SRV actuation were modeled using the input parameters specified in TSAR Table 9-1. Therefore, the NRC staff concludes that this limitation and condition is satisfied.

As required by M+ LTR Limitation and Condition 12.23.8, the plant-specific ATWS analyses must account for plant and fuel design-specific features. The ATWS analyses were performed based on GNF2 fuel designs from PBAPS, Unit 2. This core is representative for addressing any cores of GNF2 fuel for both PBAPS, Units 2 and 3. Therefore, the NRC staff concludes that this limitation and condition is satisfied.

As required by M+ LTR Limitation and Condition 12.23.3, the plant-specific ATWS analyses assumed SRV setpoints that account for a 3 percent drift tolerance. The licensee stated that the PBAPS as-found SRV lift setpoint tests do not show a propensity for setpoint drift higher than the 3 percent drift tolerance. Therefore, the NRC staff concludes that the SRV upper tolerances used in the ATWS analyses are consistent with the plant-specific performance and meets this limitation and condition.

As discussed in TSAR Section 9.3.1, the ATWS overpressure and ATWS with core instability events for PBAPS MELLLA+ were evaluated using a plant-specific TRACG model. Consistent with the requirement of Methods LTR Limitation and Condition 9.20, the void reactivity coefficients bias and uncertainties used in the latest version of TRACG are applicable to the GNF2 lattice designs loaded in the core, and thus meets this limitation and condition.

*TSAR Section 9.3.1.1 – ATWS (Overpressure) – TRACG*

As discussed in TSAR Section 9.3.1.1, TRACG ATWS Overpressure LTR (Reference 46) Limitation and Condition 4.2 requires reporting the plant-specific power-to-flow ratio at rated power and minimum core flow (CF). In addition, TRACG ATWS Overpressure LTR Limitation and Condition 4.3 requires the actual power level be stated from the TRACG ATWS application. Accordingly, the licensee reported that **[[**

**]]**. Therefore, the NRC staff concludes that these limitations and conditions are satisfied.

The NRC staff concludes that the TPO ATWS overpressure analysis is acceptable because the analysis results for the limiting events meet the acceptance criteria design limits for reactor vessel pressure, as shown in TSAR Table 9-2.

*TSAR Section 9.3.1.2 – ATWS (Suppression Pool Pressure and Temperature) – ODYN*

As discussed in TSAR Section 9.3.1, the TPO ATWS analysis was performed, in part, by using the NRC-approved ODYN code. As discussed in TSAR Section 9.3.1.2, the ATWS events analyzed by the licensee for determining the limiting (maximum) suppression pool temperature response, while considering the limiting (minimum) RHR heat removal capability, included main steam isolation valve closure (MSIVC), pressure regulator failure open (PRFO), inadvertent opening of a relief valve (IORV), and loss of offsite power (LOOP). **[[**

**]]**. In the supplement dated August 8, 2017, the licensee further stated that the LOOP event does not result in a reduction in the RHR suppression pool cooling capability relative to the MSIVC and PRFO events because the same RHR suppression pool cooling capability that is credited in the analysis of the MSIVC and PRFO events is provided during the LOOP event by equipment powered by the standby alternating current power supply.

**[[**

**]]**.

The suppression pool heat capacity temperature limit (HCTL) is defined so that transferring all the stored energy of the pressurized primary system to the suppression pool will not result in containment integrity violation. **[[**

]]. The TPO MELLLA+ end-of-cycle (EOC) results for the containment pressure and temperature (as shown in TSAR Figures 9-8c and 9-10c) are more limiting than the beginning-of-cycle (BOC) results (as shown in TSAR Figures 9-4c and 9-6c). As shown in TSAR Table 9-2, the peak containment pressure is **[[ ]]** and the peak suppression pool temperature is **[[ ]]**. The ATWS acceptance criteria for the containment pressure and temperature are the design limits of 56 psig and 180 °F, respectively.

In an RAI, the NRC staff requested that the licensee explain the basis for the acceptance criteria of the limiting temperature 180 °F of the containment structure and justify that it is conservative. The licensee was also requested to provide the HCTL at the normal suppression pool level and the SRV opening pressure. In response, in the supplement dated August 8, 2017, the licensee stated that:

> Maintaining suppression pool temperature below 180°F is applied as an acceptance criterion during an ATWS event in order to avoid damage to the HPCI pump and ensure availability of the HPCI system. As indicated in TSAR Table 9-2, the peak suppression pool temperature during an ATWS event at TPO/MELLLA+ conditions is analyzed to be less than 180°F. The limiting temperature of 180°F is conservatively lower than the 281°F design temperature of the containment structure (drywell and suppression chamber) given in Table 5.2.1 of the PBAPS UFSAR.
>
> For CLTP, the HCTL is 178.5°F at the minimum suppression pool water level of 14.5 ft that is allowed by Technical Specifications, and the lowest SRV opening pressure of 1135 psig. For TPO, under the same suppression pool water level and SRV opening pressure conditions, the HCTL will be 178.4°F for PBAPS Unit 2. The TPO HCTL for PBAPS Unit 3 will be determined as part of the core reload process and is expected to be similar to the Unit 2 limit given that both cores are comprised of all GNF2 fuel.

The NRC staff concludes that the TPO ATWS analysis for suppression pool pressure and temperature is acceptable because the licensee used the NRC-accepted ODYN code, and the analysis results for the limiting events meet the acceptance criteria design limits for containment pressure and suppression pool temperature, as shown in TSAR Table 9-2.

*TSAR Section 9.3.1.3 – ATWS (Peak Cladding Temperature) – ODYN/TASC*

As discussed in TSAR Section 9.3.1.3, for ATWS events, the acceptance criteria for PCT and local cladding oxidation for emergency core cooling systems, defined in 10 CFR 50.46, are adopted to ensure an ATWS event does not impede core cooling. The licensee stated that for TPO, the fuel PCT during an ATWS event is 1,483 °F, local cladding oxidation is less than 17 percent, and coolable geometry is ensured.

The NRC staff concludes that there is reasonable assurance that an ATWS event at PBAPS following the TPO uprate will be acceptable with respect to core cooling, since the analysis results meet the applicable criteria in 10 CFR 50.46 (i.e., PCT of less than 2,200 °F, maximum cladding oxidation less than 17 percent, and coolable core geometry).

*TSAR Section 9.3.1.4 – ATWS with Core Instability – TRACG*

As discussed in TSAR Section 9.3.1.4, a similar analysis to that of the current analysis of record for ATWS with core instability (ATWSI) was performed for TPO RTP. The licensee stated that the conclusions remain the same; a coolable geometry is maintained and the ATWS acceptance criteria remain satisfied. This result is an expected outcome because both the MELLLA+ and TPO ATWSI events initiate from the same rod line and, therefore, reduce to nearly identical power, flow, and pressure conditions following the RPT and prior to instabilities resulting in an insignificant change of 10 °F in PCT.

The initial power and CF do not directly impact an ATWSI event. The important parameters are the power, flow, and pressure conditions after a recirculation pump trip (in both turbine trip with bypass (TTWBP) and RPT ATWSI events). Because the MELLLA+ and TPO initiate from the same rod line (MELLLA+ boundary), the thermal-hydraulic conditions after the RPT will be approximately the same; therefore, the severity will also be similar.

M+ LTR Limitation and Condition 12.19 requires that a plant-specific ATWSI calculation be performed to demonstrate that emergency operating procedure actions, including boron injection and water level control strategy, effectively mitigate an ATWS event with large power oscillations in the MELLLA+ operating domain. This plant-specific analysis was performed for MELLLA+. As required, a plant-specific ATWSI analysis at TPO RTP was also performed, thus meeting this limitation and condition.

M+ LTR Limitation and Condition 12.23.5 requires that the power density be less than 52.5 MWt/Mlbm/hr. For PBAPS, the plant-specific maximum power-to-flow ratio at rated power and minimum CF is 46.0 MWt/Mlbm/hr. This value for the maximum power-to-flow ratio meets the limitation and condition.

The plant-specific TRACG calculation modeled in-channel water rod flow in accordance with M+ LTR Limitation and Condition 12.24.1. The plant-specific ATWSI calculation was performed using the latest NRC-approved neutronic and thermal-hydraulic codes TGBLA06/PANAC11 and TRACG04 (Reference 11). A GNF2 equilibrium core was used for the calculation; this complies with M+ LTR Limitation and Condition 12.3.d. The TRACG ATWSI analysis results are included in the TSAR consistent with M+ LTR Limitation and Condition 12.23.6.

[[

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In response to an NRC staff question regarding the statepoints at which the limiting ATWS results were obtained, the licensee stated in the supplement dated August 8, 2017, that limiting ATWS events for peak vessel bottom pressure, ATWS PCT, and ATWSI PCT, are all performed at the same statepoint corresponding to point J' (i.e., J prime) for MELLLA+ and point J for TPO on TSAR Figure 1-1a, "Power/Flow Map for TPO." The statepoints are the same (rated power, minimum flow) other than the slightly higher TPO power and higher core flow on the same MELLLA+ boundary. The results are as expected, and the NRC staff finds this acceptable.

Based on the above considerations, the NRC staff concludes that the TPO ATWSI analysis results are acceptable since acceptable methods were used and the applicable ATWS review criteria were met.

*TSAR Section 9.3.1.5 – SLC System Performance and Hardware*

As discussed in TSAR Section 9.3.1.5, based on the results of the plant-specific ATWS analysis, the maximum reactor lower plenum pressure following the limiting ATWS event reaches 1,207 psia during the time the SLC system is analyzed to be in operation. Compared to the results of the PBAPS analysis for MELLLA+ (1,206 psia), there is an insignificant difference in lower plenum pressure. Consequently, the pump discharge pressure and operating pressure margin for the pump discharge relief valves remain about the same. As such, the plant-specific analysis is consistent with M+ LTR Limitation and Condition 12.23.9.

Based on analyses performed using NRC-approved methodologies, the available margin for peak ATWS parameters, and the applicable limitations and conditions being met as discussed above, the NRC staff concludes that there is reasonable assurance that the SLC system has the capability to mitigate an ATWS event under TPO conditions at PBAPS.

*TSAR Section 9.3.1.6 – Suppression Pool Temperature following ATWS Event*

In Section 9.3.1.6 of the TSAR, the licensee stated:

> The boron injection rate requirement for maintaining the peak suppression pool water temperature limits, following the limiting ATWS event with SLCS [standby liquid control system] injection, is not increased for TPO. Therefore, the suppression pool temperature following an ATWS event meets all CLTR [Reference 40] dispositions.

In the supplement dated August 8, 2017, the license further stated that:

> For the proposed PBAPS TPO, the CLTR (NEDC-33004P-A, Revision 4) requires a plant-specific analysis to ensure sufficient standby liquid control system capability. The intent of TSAR Section 9.3.1.6 is to state that this was accomplished. A plant-specific analysis was performed, and Section 9.3.1.2 and Table 9-2 [of the TSAR] show that all acceptance criteria are met with the same boron injection rate as was used for CLTP/MELLLA+. Since a plant-specific analysis was performed with acceptable results, further dispositioning of the CLTR is not required.

The NRC staff concludes that the LAR is acceptable with respect to suppression pool temperature following ATWS events because the licensee has performed a plant-specific analysis with acceptable results, as shown in TSAR Table 9-2.

*TSAR Section 9.3.2 – Station Blackout*

As defined in 10 CFR 50.2, station blackout (SBO) refers to a complete loss of AC electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the loss of offsite power concurrent with a turbine trip and failure of the onsite emergency AC power system (i.e., EDGs). SBO does not include the loss of available AC power to buses fed by station batteries through inverters or the loss of power from alternate AC sources. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63.

As discussed in UFSAR Section 5.2.4.3.1, the current SBO analysis for PBAPS was evaluated at 100 percent of CLTP (i.e., at 3,951 MWt). The TPO uprate would authorize an increase in the maximum licensed RTP level from 3,951 MWt to 4,016 MWt, which is an increase of approximately 1.66 percent. In Section 9.3.2 of the TSAR, the licensee stated a plant-specific analysis was performed to confirm continued compliance to 10 CFR 50.63 at TPO RTP conditions. As discussed in the supplement dated July 13, 2017, the plant-specific analysis for TPO was performed at a bounding power level of 4,018 MWt.

The licensee provided a summary of the results of the plant-specific analysis of the SBO event for TPO in the supplement dated July 13, 2017. The results of the analysis demonstrated satisfactory response to an SBO event using the guidelines of Nuclear Management and Resource Council, Inc. (NUMARC) NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors" (Reference 48), and Regulatory Guide 1.155, "Station Blackout" (Reference 49). The licensee reviewed the adequacy of the condensate/reactor coolant inventory, the capacity of the Class 1E batteries, the SBO compressed nitrogen requirements, the effect of loss of ventilation on rooms that contain equipment essential for plant response to an SBO event, and the ability to maintain containment integrity. The licensee's assessment determined that the plant continues to show a satisfactory response to an SBO event for a coping period of 8 hours.

Based on the NRC staff's review of the above information, it concludes that the TPO uprate will have no impact on PBAPS's SBO coping duration. Therefore, the staff finds that PBAPS will continue to meet the requirements of 10 CFR 50.63 under TPO uprate conditions.

## 3.10 <u>TSAR Section 10.0 – Other Evaluations</u>

The following provides the NRC staff's technical review of the topics in Section 10.0 of the TSAR.

### 3.10.1 <u>TSAR Section 10.1 – High Energy Line Break</u>

UFSAR Appendix A, "Pressure Integrity of Piping and Equipment Pressure Parts," Section A.10.1, identifies the high energy systems based on the following criteria: the maximum operating temperature exceeds 200 °F and the maximum operating pressure exceeds 275 psig. The licensee identified the following systems in the reactor building outside the containment that contain high energy piping: main steam, FW, HPCI, RCIC, RWCU, and high energy sampling and instrument sensing lines.

The licensee stated that the high energy system operating pressure and temperature under the TPO uprate conditions remain unchanged or only change slightly; therefore, there is no significant change in the mass and energy release during a high energy line break (HELB). Since there is no change in the piping configuration, the postulated break locations remain unchanged. The insignificant change in the operating condition, if any in relation to the effect on line break calculations and the existing analyses, bound TPO uprate conditions.

TSAR Section 10.1 states that vessel dome pressure and other portions of the RCPB remain at current operating pressure or lower. In an RAI, the licensee was requested to describe the portions of the RCPB under the TPO conditions that would be operating at a lower pressure than the operating pressure at the CLTP and provide the reasons. In the supplement dated August 8, 2017, the licensee stated:

> As the main steam flow rate increases due to the TPO uprate, the frictional pressure drop increases while the reactor vessel dome pressure remains constant from CLTP to TPO. Therefore, at TPO conditions, main steam downstream of the reactor will operate at slightly lower pressures than at CLTP.

The NRC staff finds the licensee's response acceptable because with the same reactor dome pressure at TPO uprate as under the CLTP conditions, and with higher steam flow rate at the TPO uprate considering frictional losses, the RCPB pressure will be slightly less in the main steam lines downstream of the reactor.

The postulated break locations remain the same because the piping configuration does not change due to the TPO uprate. Therefore, the consequences of any postulated HELB remain bounded for the TPO uprate.

The licensee stated that at the TPO uprate, HELBs outside the containment would result in an insignificant change in the subcompartment pressure and temperature profiles. The affected building and cubicles that support safety-related functions are designed to withstand the resulting pressure and thermal loading following a HELB at the TPO uprate. In an RAI, the licensee was requested to explain why the change is insignificant compared to the current pressure and temperature profiles and the existing design margin. In the supplement dated August 8, 2017, the licensee stated:

> TSAR Section 10.1.2.1 discusses the change in feedwater mass and energy release due to TPO uprate in the main steam tunnel and states the increase is insignificant. However, the analyzed pressure and temperature profiles in the main steam tunnel due to a Main Steam Line Break (MSLB) bound those from a feedwater line break, and are unchanged for TPO. Subcompartment pressure and temperature profiles for all analyzed HELBs remain unchanged with TPO.

The NRC staff finds the licensee's response acceptable because the analyzed pressure and temperature transients from a main steam line break, which already bound the similar transients from a FW line break, are not changed from the CLTP conditions to the TPO uprate conditions.

The following is the NRC staff evaluation of the HELB analysis under TPO uprate conditions for the systems evaluated in the UFSAR, Appendix A, Section A.10.

*TSAR Section 10.1.1 – Steam Line Breaks*

In TSAR Section 10.1.1, the licensee stated:

> The critical parameter affecting the high energy steam line break analysis is the reactor vessel dome pressure. Because the operating pressure and flow restrictor remain unchanged, there is no change in steam line break flow rate. The [main steam (MS) line break] (MSLB) is used to establish the peak pressure and the temperature environment in the MS tunnel. Design margins within the HELB analysis for a MSLB provide adequate margin to the limits in the steam tunnel. With the constant pressure uprate, there is also no change in HPCI or RCIC steam line operating pressures or calculated HELB mass and energy releases with TPO uprate. Therefore, existing HELB analyses for these breaks remain bounding for TPO uprate.

The NRC staff finds the licensee's evaluation for the HELB in the MS tunnel acceptable for the TPO uprate because the reactor dome operating pressure is not changed and there are no changes in the piping configuration including the steam flow restrictor. The licensee's evaluation for the steam line breaks in the RCIC and HPCI systems is also acceptable to the NRC staff for the TPO uprate because the steam line operating pressures for these systems is not changed, and the piping configuration for these systems is not changed.

*TSAR Section 10.1.2.1 – Feedwater Line Breaks*

In TSAR Section 10.1.2.1, the licensee stated:

> The TPO uprate affects the FW temperature by < 2 °F and enthalpy by less than 2.0 [British thermal units per pound mass (BTU/lbm)], which results in an insignificant increase in FW mass and energy release. As a result of the small change in FW energy, the blowdown and energy release rate increase marginally. For small changes in FW process parameters, the feedwater line break conditions are bounded by the MSLB conditions in the MS tunnel. MSLB continues to be the bounding pipe break for the MS tunnel. Therefore, the original HELB analysis is bounding.

The NRC staff agrees that, under the TPO uprate conditions, a minor change (less than 2 °F) of the FW temperature, would result in an insignificant increase in the blowdown energy from a FW line break in the MS tunnel; the effect of which remains bounded by the MSLB in the MS tunnel.

*TSAR Section 10.1.2.2 – ECCS Line Breaks*

In TSAR Section 10.1.2.2, the licensee stated:

> ECCS liquid lines are normally isolated from the reactor during normal operations and are excluded from the PBAPS design and licensing basis for HELB. Therefore, the previous HELB analysis for breaks outside primary containment is bounding for the TPO uprate condition.

The NRC staff finds the licensee's evaluation acceptable because the ECCS liquid lines would be kept isolated during normal operation at the TPO uprate conditions.

*TSAR Section 10.1.2.3 – RCIC and HPCI System Line Breaks*

In TSAR Section 10.1.2.3, the licensee stated:

> RCIC and HPCI liquid lines are normally isolated from the reactor during normal operations and are excluded from the PBAPS design and licensing basis for HELB. Because there is no increase in the reactor dome pressure relative to the CLTP analysis, the mass and energy release for postulated RCIC and HPCI steam line breaks does not increase. Therefore, the previous HELB analysis is bounding for the TPO uprate conditions.

The NRC staff agrees that the mass and energy release from RCIC and HPCI steam line breaks is not affected from the current analysis because during operation, these systems draw steam from the reactor dome whose pressure is not changed from the CLTP conditions. Moreover, these systems are normally kept isolated and are only operated for surveillance testing during normal plant operation. For the same reason, the RCIC and HPCI liquid line break analysis is not affected because these lines are kept isolated during normal operation, and the mass and energy release during a break while these systems are under surveillance testing does not change from the CLTP mass and energy release.

*TSAR Section 10.1.2.4 – RWCU System Line Breaks*

In TSAR Section 10.1.2.4, the licensee stated:

> The existing design basis calculations bound TPO uprate conditions for evaluating the blowdown rate and energy release rate; therefore, the current HELB analyses bound the TPO uprate.

UFSAR Appendix A, Section A.10.7.5, describes the current high energy line break evaluation of the portion of the RWCU system outside the containment. The NRC staff agrees that the TPO uprate does not affect the current analysis because of the following reasons: (1) the system operating conditions are not changed; (2) the piping configuration is not changed; and (3) the licensee stated that the current design-basis calculations bound the TPO uprate conditions for the HELB blowdown mass and energy release rates.

*TSAR Section 10.1.2.5 – CRD System Line Breaks*

In TSAR (Reference 2) Section 10.1.2.5, the licensee stated:

> The CRD [control rod drive] system and supporting equipment operation are not affected by a TPO uprate. CRD is not considered to be a high energy system and is excluded from HELB analysis per the PBAPS design and licensing basis. Therefore, CRD lines are not affected.

The NRC staff accepts that the CRD system is excluded from HELB analysis under the TPO uprate conditions because of the following reasons: (1) in the current licensing basis UFSAR

Section A.10, the CRD system is not identified as a high energy system, and (2) the licensee has not identified any changes in the CRD system piping design conditions or its configuration for the TPO uprate.

*TSAR Section 10.1.2.6 – Building Heating and Auxiliary Steam Line Breaks*

In TSAR Section 10.1.2.6, the licensee stated:

> Building heating and auxiliary steam systems are not considered to be high energy systems and are excluded from HELB analysis per the PBAPS design and licensing basis (Reference [UFSAR Section A.10]). Therefore, building heating and auxiliary steam lines are not affected.

The NRC staff accepts that the building heating and auxiliary systems are excluded from HELB analysis under the TPO uprate conditions because of the following reasons: (1) in the current licensing basis described in UFSAR Section A.10, these systems are not identified as a high energy system, and (2) the licensee has not identified any changes in these systems' piping design conditions or their configuration for the TPO uprate.

*TSAR Section 10.1.2.7 – Pipe Whip and Jet Impingement*

In TSAR Section 10.1.2.7, the licensee stated:

> Because there is no change in the nominal vessel dome pressure, pipe whip and jet impingement loads do not significantly change. Existing calculations supporting the dispositions of potential targets of pipe whip and jet impingement from postulated HELBs bound the safe shutdown effects at the TPO uprate conditions. Existing pipe whip restraints, jet impingement shields, and their supporting structures are also adequate for the TPO uprate conditions.

UFSAR Section A.10.2 describes the criteria for consideration of the effects of a piping system break outside containment, including the effects of pipe whip and jet impingement. Based on the following reasons, the NRC staff accepts that the TPO uprate will not affect the current pipe whip and jet impingement analysis: (1) the reactor dome pressure is not changed; therefore, the pipe whip and jet impingement loads from the breaks in the reactor coolant piping will not change; and (2) the licensee stated that the current calculations from postulated HELBs bound the effects during safe shutdown at the TPO uprate conditions. For the same reasons, the NRC staff agrees that the TPO uprate does not affect the pipe whip restraints, jet impingement shields, and their supporting structures.

*TSAR Section 10.1.2.8 – High Energy Sampling and Instrument Line Breaks*

In TSAR Section 10.1.2.8, the licensee stated:

> High energy sampling and instrument lines are determined to not be the limiting breaks at TPO uprate conditions. Therefore, high energy sampling and instrument line breaks are not affected.

UFSAR Section A.10.7.6 describes the current licensing basis for the breaks in the high energy sampling and instrument sensing lines. The NRC accepts that HELB analysis is not affected due to breaks in these lines for the following reasons: (1) there is no change in the configuration of operating conditions in these lines for the TPO uprate; and (2) due to their small diameter, the mass and energy release from the breaks in these lines would be less limiting than the mass and energy release from the larger high energy lines in the same area.

*TSAR Section 10.1.2.9 – Internal Flooding from HELB*

Since the TPO uprate does not change the reactor dome pressure, the licensee's current evaluation of the flooding zones that contain a postulated HELB in a system carrying the reactor coolant remains unchanged. Therefore, the NRC staff concludes that the high energy line systems operational modes, plant internal flooding analysis, and safe shutdown analysis evaluated for HELB are not affected by the TPO uprate.

3.10.2 TSAR Section 10.2 – Moderate Energy Line Break

NUREG-0800, "Standard Review Plan, Branch Technical Position 3-3, 'Position against Postulated Piping Failures in Fluid Systems Outside Containment,'" Revision 3 (ADAMS Accession No. ML070800027), Appendix A, defines moderate energy fluid systems as the fluid systems that during normal plant conditions are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following conditions are met: maximum operating temperature is 200 °F or less and maximum operating pressure is 275 psig or less.

As discussed in TSAR Section 10.2, the following systems in the reactor building outside the primary containment contain moderate energy piping: control rod hydraulic, residual heat removal, SLC, reactor core isolation cooling, core spray, instrument nitrogen, fuel pool cooling, post-accident sampling, high-pressure coolant injection, high-pressure service water, emergency service water, reactor building cooling water, service air, instrument air, fire water, domestic water, demineralized water, chilled water, and radiation monitoring.

The licensee evaluated the plant flooding zones and stated that none of these zones contains a potential moderate energy line break that is affected by the TPO uprate. The NRC staff agrees with the licensee that the TPO uprate has no effect on the currently identified potential adverse effects of breaks in moderate energy lines and that the current plant flooding analysis is unaffected because of the following: (1) the TPO uprate does change the configuration or identify any new moderate energy piping, (2) the sources of moderate energy flooding and protection requirements for safe shutdown equipment for a postulated moderate energy line break are not dependent on power level, and (3) the flooding sources are negligibly affected and there in no change in the protection requirements.

3.10.3 TSAR Section 10.3 – Environmental Qualification

In TSAR Section 10.3, the licensee stated that an evaluation of environmental qualification (EQ) parameters such as temperature, pressure, and radiation was performed to evaluate whether or not any potential parameter would change due to the TPO uprate. The licensee stated that the environmental conditions for safety-related electrical equipment were reviewed to ensure that the existing qualification for the normal and accident conditions expected in the area where the electrical equipment resides remains adequate.

In Section 10.3.1.1 of the TSAR, the licensee stated that EQ for safety-related electrical equipment located inside the containment is based on DBA-LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. The licensee stated that current accident conditions for temperature and pressure are based on analyses initiated from at least 102 percent of CLTP. Normal temperatures may increase slightly near the feedwater and reactor recirculation lines and will be evaluated through the EQ temperature monitoring program, which tracks such information for equipment aging considerations. The current radiation levels under normal plant conditions also increase slightly under TPO conditions. However, the current plant environmental envelope for radiation is not exceeded by the changes resulting from the TPO uprate.

In Section 10.3.1.2 of the TSAR, the licensee stated that accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from a main steam line break in the steam tunnel or other high energy line breaks (HELBs), whichever is limiting for each area. The licensee stated that the existing pressure and temperature profiles bound the TPO uprate conditions. The licensee also determined that the current plant environmental envelope for radiation is not exceeded by the changes resulting from the TPO uprate.

In the supplement dated July 13, 2017, the licensee stated that the TPO uprate does not increase the normal vessel dome pressure; therefore, it will have a negligible effect on normal pressure and temperature experienced by equipment outside containment, as well as normal pressure inside containment. The licensee stated that the pressure inside containment and the pressure and temperature outside containment at TPO conditions will remain bounded by the existing environmental parameters specified for use in the EQ program.

In the supplement dated July 13, 2017, the licensee stated that temperature, pressure, and humidity values for normal operating conditions are maintained by the normal HVAC systems and will remain within the existing parameters specified for use in the EQ program. The licensee also stated that the qualification for normal humidity conditions is bounded by the qualification for accident conditions. For accident conditions, the equipment inside and outside containment is qualified to a 100 percent relative humidity level and, therefore, the qualification is bounding of TPO conditions.

Based on the NRC staff's review of the above information, the NRC staff concludes that the TPO uprate will have no impact on EQ of the PBAPS electrical equipment due to the fact that the existing qualification for the normal and accident conditions for electrical equipment inside and outside containment remain adequate and bounded by the TPO uprate conditions. Therefore, the NRC staff concludes that PBAPS will continue to meet the requirements of 10 CFR 50.49 under power uprate conditions.

3.10.4 TSAR Section 10.4 – Testing

As discussed in TLTR Section 5.11.9, guidelines for the TPO uprate power ascension testing plan are provided in TLTR, Appendix L, Section L.2. In TSAR Section 10.4, the licensee stated that the PBAPS power ascension test plan is based on the guidelines in the TLTR. Some of the key attributes of the planned testing for PBAPS include the following:

- Routine measurements of reactor and system pressures, flows, and vibration of selected major rotating equipment will be taken near 95 percent CLTP, 100 percent CLTP, and at full TPO RTP conditions.

- Demonstration of acceptable fuel thermal margin will be performed prior to and during power ascension.

- Performance of the pressure and feedwater/level control system will be recorded during power ascension. The checks will utilize the methods and criteria described in the original startup testing of these systems to demonstrate acceptable operational capability.

The NRC staff has reviewed the proposed testing in Section 10.4 of the TSAR and finds that it is consistent with the generic guidelines in TLTR, Appendix L, Section L.2.

The licensee also stated that large transient testing is not necessary since the increase in power for the TPO uprate is small and that testing during initial plant startup demonstrated the adequacy of the safety and protection systems for such large transients. The licensee also stated that operational occurrences have shown that the plant response is bounded by the safety analyses for these events. The NRC staff finds that the licensee's justification for not performing large transient tests is consistent with the generic justification provided in TLTR, Appendix L, Section L.2.

The NRC staff concludes that the LAR is acceptable with respect to testing based on the findings discussed above.

3.10.5 <u>TSAR Section 10.5 – Operator Training and Human Factors</u>

As discussed in SE Section 1.2, RIS 2002-03 provides guidance on the scope and detail of the information that should be provided to the NRC staff for MUR power uprate LARs. Attachment 1 to the RIS lists the specific topic areas that should be addressed by the licensee and the level of detail for each topic area.

Attachment 4 to the licensee's application dated February 17, 2017, provides a cross-reference to the topic areas discussed in Attachment 1 to RIS 2002-03. Items VII.1 through VII.4 in Attachment 1 to the RIS pertain to the operator training and human factors review. The areas associated with these items are evaluated below.

*Operator Actions*

As discussed in TSAR Section 10.5, the licensee stated that the operator actions for maintaining safe shutdown, core cooling, and containment cooling do not change for the proposed MUR power uprate. The licensee reviewed the following safety analyses for potential impact for operator actions: (1) containment system performance (TSAR Section 4.1), (2) fire protection (TSAR Section 6.7), and (3) special events, specifically, anticipated transient without scram (ATWS) and station blackout (TSAR Section 9.3). For containment isolation, the proposed change will not affect the ability of containment isolation valves and operators to perform their required functions because previous evaluations performed by PBAPS were performed at 102 percent of the CLTP level of 3,951 MWt. Therefore, containment system performance is bounded by the current evaluation for the proposed MUR power uprate. Fire protection is not compromised because operation at the proposed CLTP of 4,016 MWt does not affect the fire suppression or detection systems. Operator actions to mitigate the consequences of a fire are not affected. TSAR Section 6.7.1 notes that there is a slight reduction in the time from event initiation to the point at which manual initiation of RPV depressurization from the control room is required for one fire safe shutdown method. However, the time for this method is still more than that available for other fire safe shutdown methods that manually initiate an RPV depressurization, and the timeline to perform the manual action is not affected.

As described in TSAR Section 9.3.1, additional analyses for the higher proposed RTP value were completed to determine any impacts to limiting ATWS events. The plant-specific analyses confirmed that no changes are required to assumed operator actions or response times for ATWS events. Similarly, TSAR Section 9.3.2 stated that additional plant-specific analyses confirmed that PBAPS will continue to meet the requirements for a station blackout event in compliance with 10 CFR 50.63. TSAR Section 10.5 also confirms that operator response to special events (i.e., ATWS and station blackout) is not affected except for the negligible impact on the timeline for operator actions to mitigate the consequences of a fire as described above.

There are non-time-critical operator actions that are associated with the LEFM system. The control room operators are alerted with a visual alarm from the plant monitoring computer when the LEFM system indicates an off-normal status. Per the Bases for the proposed new TRM Section 3.20 (Attachment 3 to the licensee's application), the alarm does not require an immediate action from the operators. If there is a sustained loss of function, the operators will have to enter the compensatory measures to reduce core thermal power specified in the TRM and the core thermal power calculations will automatically revert back to the feedwater flow venturis that remain calibrated.

The NRC staff finds that the information provided by the licensee adequately addresses the guidance in Section VII, Item 1 of Attachment 1 to RIS 2002-03. Based on the information provided, the NRC staff concludes that the proposed MUR power uprate will not adversely impact the licensee-identified operator actions or their response times.

*Emergency and Abnormal Operating Procedures*

As discussed in TSAR Section 10.9, the emergency operating procedure (EOP) action thresholds are plant unique and will be addressed using standard procedure updating processes. The licensee expects that the MUR power uprate will involve only minor changes to the operator action thresholds and will have no effect on the EOP strategies.

In the supplement dated July 13, 2017, the licensee stated that the MUR update will have no effect on the abnormal operating procedure (AOP) strategies with only minor changes to the setpoints and thresholds in the procedures.

The NRC staff finds that the information provided by the licensee adequately addresses the guidance in Section VII, Item 2.A of Attachment 1 to RIS 2002-003. Based on the information provided, the NRC staff concludes that the proposed changes to EOPs and AOPs do not adversely affect defense-in-depth or safety margins.

*Changes to Control Room Controls, Displays, and Alarms*

The licensee stated in Section 3.3.2 of Attachment 1 to the dated February 17, 2017, that installation of the LEFM ultrasonic flow measurement instrumentation was completed at PBAPS in 2003 in conjunction with an initial MUR. Therefore, there are no changes to plant physical instrumentation or displays associated with the proposed MUR power uprate. Per the Bases for the proposed new TRM Section 3.20, the currently installed feedwater flow venturis will remain available and used if the LEFM system is not functional. The LEFM system computer converts the flow meter data into feedwater flow and temperature signals that provide a self-check and determination of flow measurement uncertainty within the plant monitoring system alarms. Per Section 3.3.4 of Attachment 1 to the licensee's application, the control room operators are promptly alerted by the plant monitoring system alarm in the control room if the LEFM system or any portion of the LEFM system becomes non-functional. Per TRM Section 3.20, the alarm does not require an immediate action from the operators. If there is a sustained loss of function, the operators will have to enter the compensatory measures to reduce core thermal power specified in the TRM, and the core thermal power calculations will automatically revert back to the feedwater flow venturis that remains calibrated. Per Section 3.3.6 of Attachment 1 to the licensee's application, the licensee's existing procedures provide guidance for controlling and maintaining reactor power within the requirements of the PBAPS operating license.

The NRC staff finds that the information provided by the licensee adequately addresses the guidance in Section VII, Item 2.B of Attachment 1 to RIS 2002-003. Based on the information provided, the NRC staff concludes that the proposed changes to the control room controls, displays, and alarms do not adversely affect defense-in-depth or safety margins.

*Control Room Plant Reference Simulator*

The licensee stated in Section 3.5.6 of Attachment 1 to the application dated February 17, 2017, that the PBAPS simulator will be modified to reflect any changes to the control room in accordance with established PBAPS certification procedures. The licensee stated that the established PBAPS plant simulator certification testing procedures satisfy the guidelines for simulator testing, performance, fidelity, and configuration control specified by ANSI/ANS-3.5-1998, "Nuclear Power Plant Simulation Facilities." This standard is endorsed by the NRC in Regulatory Guide 1.149, Revision 3. In the supplement dated July 13, 2017, the licensee stated that modifications to the simulator will be limited to model changes to reflect the conditions at the higher MUR power level, setpoint changes, plant computer changes such as alarm points, and minor instrumentation changes to provide adequate indicating range.

The NRC staff finds that the information provided by the licensee adequately addresses the guidance in Section VII, Item 2.C of Attachment 1 to RIS 2002-003. Based on the information provided, the NRC staff concludes that the proposed changes to the control room plant reference simulator do not adversely affect defense-in-depth or safety margins.

*Operator Training*

As discussed in TSAR Section 10.5, the licensee stated that no additional training is required for the proposed MUR power uprate apart from normal training for plant changes. For MUR power uprate conditions, operator response to transient, accident, and special events is not affected except for the negligible impact on the timeline for operator actions to mitigate the consequences of a fire as described above. Operator actions for maintaining safe shutdown, core cooling, and containment cooling do not change with these proposed amendments.

Per Section 3.5.6 of Attachment 1 to the licensee's application, operator training will be completed prior to implementation of the proposed MUR power uprate in accordance with the PBAPS plant training and simulator program.

The NRC staff finds that the information provided by the licensee adequately addresses the guidance in Section VII, Item 2.D of Attachment 1 to RIS 2002-003. Based on the information provided, the staff concludes that the proposed changes to the operator training program do not adversely affect defense-in-depth or safety margins.

*Modifications*

The licensee has stated in Section 3.5.3 of Attachment 1 to the application dated February 17, 2017, that there are no modifications required to plant systems to support the proposed amendments other than a limited number of instrument setpoint changes, rescaling, or replacements. These instrument setpoint modifications are integral to the implementation of the proposed power uprate. The licensee also noted that hardware changes to the turbine may be required to achieve the full advantage of the proposed MUR power uprate. However, these modifications are an economic decision and the licensee did not request approval of the modifications as part of the LAR. The Commissions' regulation at 10 CFR 50.59 provides the criteria that the licensee will use to determine if prior NRC approval is needed to implement the modifications.

The licensee stated in TSAR Section 5.2.1 that the performance of the pressure control system was analyzed at a TPO bounding power level 4,018 MWt. The analysis confirmed that no modifications are required for the pressure regulation system or steam bypass valve system to accommodate the proposed MUR power uprate. In addition, no modifications are necessary to the operator interface indications, controls, or alarm annunciators provided in the main control room.

As noted above, operator training will be completed prior to implementation of the proposed MUR power uprate in accordance with the PBAPS plant training and simulator program. The NRC staff finds that the information provided by the licensee adequately addresses the guidance in Section VII, Item 3 of Attachment 1 to RIS 2002-003.

*Procedure Revisions – Licensed Power Level*

The licensee stated the following in Section 3.3.6 of Attachment 1 to the application dated February 17, 2017:

> PBAPS Unit 2 and Unit 3 have procedures that provide guidance for monitoring and controlling reactor power and ensuring that reactor power remains within the requirements of the operating license.

The NRC staff finds that the information provided by the licensee adequately addresses the guidance in Section VII, Item 4 of Attachment 1 to RIS 2002-003.

*Operator Training and Human Factors Conclusion*

Consistent with the discussion above, the NRC staff finds that the information provided by the licensee adequately addresses the guidance in Items VII.1 through VII.4 in Attachment 1 to RIS 2002-003. Based on review of this information, the staff concludes that the LAR is acceptable with respect to operator training and human factors.

3.10.6 TSAR Section 10.6 – Plant Life

As discussed in TSAR Section 10.6, the licensee evaluated the effect of the proposed TPO uprate on age-related degradation of plant equipment. The licensee concluded that the longevity of most equipment will not be affected by TPO because there is no significant change in the operating conditions, and any changes in operating conditions would be bounded by a previous evaluation completed for EPU at 102 percent of CLTP. The licensee also stated that no additional maintenance, inspection, testing, or surveillance procedures are necessary for the small change being introduced by the proposed TPO uprate.

The NRC renewed the operating licenses for PBAPS, Units 2 and 3, on May 7, 2003 (ADAMS Accession No. ML031150073). Appendix Q to the PBAPS UFSAR contains the summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation as required by 10 CFR 54.21(d). As indicated in Appendix Q, plant operation under EPU conditions has been incorporated into this appendix.

The NRC staff agrees that with respect to any age-related degradation, no additional maintenance, inspection, testing, or surveillance procedures are necessary because the changes in operating conditions are not significant. In addition, for plants operating pursuant to a renewed license (such as PBAPS, Units 2 and 3), those few components that might be affected already have effective plant programs in place to detect and mitigate age-related degradation. In addition, since: (1) the time-limited aging analyses for PBAPS, Units 2 and 3, described in UFSAR Appendix Q, account for operation under EPU conditions; and (2) any changes in operating conditions would be bounded by the previous evaluation completed for EPU at 102 percent of CLTP, there is reasonable assurance the time-limiting aging analyses are adequate to ensure the affected SSCs will continue to perform their intended functions under TPO uprate conditions.

Based on the above considerations, the NRC staff concludes that the LAR is acceptable with respect to this topic.

3.10.7 TSAR Section 10.7 – NRC and Industry Communications

As discussed in TLTR, Appendix B, Section B.4, a plant-specific review of generic NRC and industry communications is not needed for a TPO uprate. Accordingly, as discussed in TSAR Section 10.7, the licensee stated that no information is required in this area.

The NRC staff concludes that the LAR is acceptable with respect to this topic since the disposition is consistent with the TLTR.

3.10.8 TSAR Section 10.8 – Plant Procedures and Programs

As discussed in TSAR Section 10.8, the licensee stated that PBAPS has previously implemented a TPO uprate, including procedure and program requirements. Therefore, only minor changes are required to restore the TPO requirements in the plant procedures and programs.

The NRC staff concludes that the normal licensee process for updating procedures (i.e., consistent with the provisions in 10 CFR 50.59) is sufficient with respect to addressing the impact of the TPO uprate on the plant procedures and programs.

3.10.9 TSAR Section 10.9 – Emergency Operating Procedures

As discussed in TLTR Section 5.11.3, the emergency operating procedure (EOP) action steps are unchanged due to a TPO uprate because the procedures are symptom-based, independent of reactor power level. However, certain threshold values for initiating mitigating actions are dependent upon power or decay heat levels. The TLTR further states that the EOP action thresholds are plant unique and will be addressed, as needed by the utility, using standard procedure processes as done in previous BWR power uprates. Consistent with the TLTR, TSAR Section 10.9 states that the TPO uprate for PBAPS, Units 2 and 3, will have no effect on EOP strategies and only minor changes to operator action thresholds. The TSAR also states that standard procedure updating processes would be used to update the EOPs.

The NRC staff concludes that the normal licensee process for updating procedures (i.e., consistent with the provisions in 10 CFR 50.59) is sufficient with respect to addressing the impact of the TPO uprate on the EOPs.

3.10.10 TSAR Section 10.10 – Individual Plant Examination

As discussed in TLTR Section 5.11.11, the change in plant risk from a TPO uprate is insignificant. Therefore, the TLTR concluded that the plant-specific individual plant examination (IPE) does not need to be updated to support the TPO uprate.

The licensee stated in TSAR Section 10.10 that the PBAPS IPE probabilistic risk assessment model and analysis will not specifically be updated for the TPO uprate because the change in risk is insignificant.

The NRC staff concludes that the LAR is acceptable with respect to this topic since the disposition is consistent with the TLTR.

3.11 License and Technical Specification Changes

The following provides the NRC staff evaluation of the proposed Renewed Facility Operating License (RFOL) and Technical Specification (TS) changes associated with the LAR. The proposed changes to the RFOL and TSs are discussed in Section 2.0 of Attachment 1 to the application, and a markup of the changes is shown in Attachment 2 to the licensee's application.

3.11.1 <u>RFOL Paragraph 2.C(1) – Maximum Power Level</u>

RFOL paragraph 2.C(1), "Maximum Power Level," would change the maximum authorized steady state reactor core power level from the CLTP level of 3,951 MWt to the proposed TPO power level of 4,016 MWt. This change reflects the proposed approximate 1.66 percent increase in thermal power level and is consistent with the licensee's supporting safety analyses. Therefore, the NRC staff concludes that the proposed change to RFOL paragraph 2.C(1) is acceptable.

3.11.2 <u>TS 1.1 – Definitions</u>

The TS 1.1 Definition for "rated thermal power" would be revised to change the RTP level from the CLTP level of 3,951 MWt to the proposed TPO power level of 4,016 MWt. This change reflects the proposed approximate 1.66 percent increase in thermal power level and is consistent with the licensee's supporting safety analyses. Therefore, the NRC staff concludes that the proposed change to TS 1.1 is acceptable.

3.11.3 <u>TS 2.1.1 – Reactor Core SLs [Safety Limits]</u>

TS 2.1.1 currently provides the following reactor core safety limit when the reactor steam dome pressure is less than 700 psia or the core flow is less than 10 percent of rated core flow:

> THERMAL POWER shall be ≤ 23% RTP.

The proposed amendment would change the safety limit to read as follows:

> THERMAL POWER shall be ≤ 22.6% RTP.

In accordance with TS 2.1.1, the current thermal limits monitoring threshold is 23.0 percent of the CLTP level of 3,951 MWt. As discussed in TSAR Sections 2.1.3 and 2.4.2, for a power-uprated plant, the thermal limits monitoring threshold may be scaled to a lower percent value to maintain the same MWt value. Therefore, for the proposed TPO uprate from the CLTP level of 3,951 MWt to the proposed TPO power level of 4,016 MWt, the thermal limits monitoring threshold can be calculated as follows:

$$\text{Thermal Limits Monitoring Threshold} = 23.0\% \times (3{,}951 \text{ MWt} / 4{,}016 \text{ MWt}) \approx 22.63\%$$

The licensee has rounded this calculated value down to 22.6 percent of the proposed TPO power level of 4,016 MWt. This represents a thermal limits monitoring threshold at a power level of 907.616 MWt (i.e., 0.226 x 4,016). The current threshold is established at a power level of 908.73 MWt (i.e., 0.23 x 3,951). As such, the NRC staff concludes that proposed change is acceptable since it conservatively establishes the thermal limits monitoring threshold at a value slightly less than the current value in terms of absolute thermal power.

3.11.4 <u>TS 3.2.1 – Average Planar Linear Heat Generation Rate (APLHGR)</u>

The TS 3.2.1 Limiting Condition for Operation (LCO) Applicability, LCO Required Action B.1, and SR 3.2.1.1 Frequency include requirements associated with a thermal power limit of 23 percent.

The proposed amendment would change the “23% RTP” value to “22.6% RTP” in all three instances.

As discussed in Section 3.4 of Attachment 1 to the licensee’s application, these proposed TS changes relate to the scaling of the thermal limits monitoring threshold. The NRC staff’s review of the scaling of the thermal limits monitoring threshold is discussed in SE Section 3.11.3.

Based on the discussion in SE Section 3.11.3, the NRC staff concludes that the proposed changes to TS 3.2.1 are acceptable.

3.11.5 <u>TS 3.2.2 – Minimum Critical Power Ratio (MCPR)</u>

The TS 3.2.2 LCO Applicability, LCO Required Action B.1, and SR 3.2.2.1 Frequency include requirements associated with a thermal power limit of 23 percent. The proposed amendments would change the “23% RTP” value to “22.6% RTP” in all three instances.

As discussed in Section 3.4 of Attachment 1 to the licensee’s application, these proposed TS changes relate to the scaling of the thermal limits monitoring threshold. The NRC staff’s review of the scaling of the thermal limits monitoring threshold is discussed in SE Section 3.11.3.

Based on the discussion in SE Section 3.11.3, the NRC staff concludes that the proposed changes to TS 3.2.2 are acceptable.

3.11.6 <u>TS 3.2.3 – Linear Heat Generation Rate (LHGR)</u>

The TS 3.2.3 LCO Applicability, LCO Required Action B.1, and SR 3.2.3.1 Frequency include requirements associated with a thermal power limit of 23 percent. The proposed amendment would change the “23% RTP” value to “22.6% RTP” in all three instances.

As discussed in Section 3.4 of Attachment 1 to the licensee’s application, these proposed TS changes relate to the scaling of the thermal limits monitoring threshold. The NRC staff’s review of the scaling of the thermal limits monitoring threshold is discussed in SE Section 3.11.3.

Based on the discussion in SE Section 3.11.3, the NRC staff concludes that the proposed changes to TS 3.2.3 are acceptable.

3.11.7 <u>TS 3.3.1.1 – Reactor Protection System (RPS) Instrumentation</u>

*Changes Related to the Turbine First Stage Pressure Function*

The following requirements in TS 3.3.1.1 relate to the turbine first stage pressure function:

- LCO Required Action E.1
- SR 3.3.1.1.13
- Table 3.3.1.1-1, Function 8, Turbine Stop Valve-Closure
- Table 3.3.1.1-1, Function 9, Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

Each of the above requirements currently specifies a value of “26.7% RTP” as a thermal power level at which the requirement applies or a thermal power level at which the plant must be reduced to under certain conditions. The proposed amendments would change the 26.7% RTP value for each of the above TSs to "26.3% RTP."

As discussed in TLTR Section F.4.2.3 and TSAR Section 5.3.1.16, the turbine first-stage pressure setpoint is used to reduce scrams and recirculation pump trips at low power levels where the turbine steam bypass system is effective. The setpoint is chosen to allow operational margin so that scrams can be avoided during turbine-generator trips at low power. The turbine first stage pressure setpoint does not change in terms of absolute power. As such, it is rescaled as follows:

Turbine First Stage Pressure setpoint = 26.7% x (3,951 MWt / 4,016 MWt) ≈ 26.3%

The NRC staff concludes that the new turbine first stage pressure value of "26.3% RTP" specified in TS 3.3.1.1 is acceptable since it maintains the setpoint at the same absolute thermal power level currently used in the PBAPS transient analyses.

*Changes Related to the Oscillation Power Range Monitor*

The proposed amendments would make two changes to TS 3.3.1.1 related to the OPRM. The specific TS requirements are as follows:

- Table 3.3.1.1-1, Function 2.f, OPRM Upscale
- LCO Required Action K.1

TS Table 3.3.1.1-1, Function 2.f addresses the power level applicable for when the OPRM Upscale function needs to be operable. As shown in Attachment 2 to the application, the licensee requested that the power level for Function 2.f be changed from "≥ 18% RTP" to "≥ 17.6% RTP." The evaluation of this proposed change is discussed below.

TS 3.3.1.1 Required Action K.1 requires that thermal power be reduced to a power level at which the OPRM Upscale function is not applicable. As shown in Attachment 2 to the application, the licensee requested that the power level be changed from "< 18% RTP" to "< 17.6% RTP." The evaluation of this proposed change is discussed below.

As discussed in the PBAPS UFSAR Section 7.5.7 and the PBAPS TS Bases, the average range power monitor (APRM) channels provide the primary indication of neutron flux within the core. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. Each APRM also includes an OPRM Upscale function that monitors small groups of LPRM signals to detect thermal-hydraulic instabilities.

The OPRM is designed to detect the onset of reactor core power oscillations resulting from thermal-hydraulic instability and suppress them by initiating a reactor scram via the reactor protection system trip logic. The OPRM Upscale function provides protection of the minimum critical power ratio (MCPR) safety limit. The OPRM system may only cause a scram when plant operation is in the Armed Region. As discussed in SE Section 3.2.4, the power level for the Armed Region corresponds to the thermal limits monitoring threshold. For TPO, the rescaled thermal limits monitoring threshold is 22.6 percent of the TPO power level as evaluated in SE Section 3.11.3.

As discussed in SE Section 3.2.4, PBAPS is currently operating under the detect and suppress solution - confirmation density (DSS-CD) long-term stability solution, consistent with the DSS-CD LTR (Reference 10), including any applicable limitations and conditions. The DSS-CD solution monitors OPRM signals to determine when a reactor scram is required. The OPRM signal is evaluated by the DSS-CD stability algorithms to determine when the signal is becoming sufficiently periodic and large to warrant a reactor scram to disrupt the oscillation.

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. The LTR states that the alternate method is acceptable because system operability is assured prior to entry into the Armed Region. Accordingly, as discussed in TSAR Section 2.4.3, the licensee stated that the minimum power level at which the OPRM should be confirmed operable is 17.6 percent of the TPO RTP level. The 17.6 percent RTP value provides a 5 percent absolute power separation (i.e., 22.6 percent – 17.6) between the OPRM Armed Region power boundary and the power at which the OPRM system should be confirmed operable.

The NRC staff concludes that the proposed changes to TS Table 3.3.1.1-1, Function 2.f and TS 3.3.1.1, Required Action K.1 are acceptable since the 17.6 percent RTP value has been established consistent with the considerations discussed above based in Section 3.5 of the NRC-approved DSS-CD LTR.

*Changes Related to the APRM Calibration*

As discussed in the TS Bases, to ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. SR 3.3.1.1.2 currently requires this calibration be done when thermal power is "≥ 23% RTP." The proposed amendments would change the "23% RTP" value to "22.6% RTP" in two places in this SR.

As discussed in Section 2.2 of Attachment 1 to the application, these proposed changes are based on rescaling of the thermal limits monitoring threshold. The NRC staff's review of the scaling of the thermal limits monitoring threshold is discussed in SE Section 3.11.3.

Based on the discussion in SE Section 3.11.3, the NRC staff concludes that the proposed changes to SR 3.3.1.1.2 are acceptable.

*Changes Related to APRM Simulated Thermal Power – High Allowable Value*

Table 3.3.1.1-1, Function 2.b, and its associated Table 3.3.1.1-1, note (b) contain requirements for the APRM Simulated Thermal Power - High function. This function, also referred to as the APRM flow-biased scram, currently has the following allowable values (AVs) as shown in the TS Table 3.3.1.1-1:

| | |
|---|---|
| Two loop operation: | $\leq 0.61\ W + 67.1\%$ RTP |
| Single loop operation: | $0.55\ (W - \Delta W) + 61.5\%$ RTP |

As shown in UFSAR Table 7.5.4, "Average Power Range Monitor Trips," "W" is the recirculation loop flow rate in percent of the design rating and "ΔW" is the difference between two loop and single loop recirculation loop flow at the same core flow.

The proposed amendments would revise the AVs to be as follows:

| | |
|---|---|
| Two loop operation: | ≤ 0.60 W + 65.9% RTP |
| Single loop operation: | 0.54 (W - ΔW) + 60.3% RTP |

As discussed in TSAR Section 5.3.7, the simulated thermal power APRM design limits for both two loop operation and single loop operation are unchanged in units of absolute core thermal power versus recirculation drive flow. As such, the associated design limits are being decreased in proportion to the TPO power uprate. As discussed in the TSAR Section 5.3 and in Section 3.5.4 of Attachment 1 to the application, the AVs are established from the design limits based on NRC-approved General Electric Topical Report NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996 (ADAMS Accession No. ML072950103, non-publicly available).

The NRC staff concludes that the proposed changes to TS Table 3.3.1.1-1, Function 2.b and TS Table 3.3.1.1-1, note (b) are acceptable since the AVs were determined based on an NRC-approved setpoint methodology.

3.11.8 TS 3.3.2.2 – Feedwater and Main Turbine High Water Level Trip Instrumentation

The TS 3.3.2.2 LCO Applicability and LCO Required Action C.2 currently include requirements associated with a thermal power limit of 23 percent. The proposed amendments would change the "23% RTP" value to "22.6% RTP" in both instances.

As discussed in the TS Bases, the feedwater and main turbine high water level trip instrumentation are required to be operable above the specified thermal power level (i.e., currently "23% RTP") to ensure that the fuel cladding integrity safety limit and the cladding 1 percent plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.3, "Linear Heat Generation Rate (LHGR)," and LCO 3.2.2, "Minimum Critical Power Ratio (MCPR)," sufficient margin to these limits exists below the specified thermal power level; therefore, these requirements are only necessary when operating at or above this power level.

As discussed in SE Sections 3.11.5 and 3.11.6, the proposed new thermal power limit applicable for the MCPR and LHGR LCOs (i.e., "22.6% RTP") is based on the scaling of the thermal limits monitoring threshold. Consistent with the TS Bases discussion above, the proposed changes to TS 3.3.2.2 are based on the scaling of the thermal limits monitoring threshold. The NRC staff's review of the scaling of the thermal limits monitoring threshold is discussed in SE Section 3.11.3.

Based on the discussion in SE Section 3.11.3, the NRC staff concludes that the proposed changes to TS 3.3.2.2 are acceptable.

3.11.9 TS 3.3.4.2 – End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

The following requirements in TS 3.3.4.2 relate to the turbine first stage pressure function:

- LCO Applicability
- LCO Required Action C.2
- SR 3.3.4.2.4

Each of the above requirements currently specifies a value of "26.7% RTP" (e.g., as a thermal power level at which the requirement applies or a thermal power level the plant must be reduced to under certain conditions). The proposed amendments would change the "26.7% RTP" value for each of the above TSs to "26.3% RTP."

Based on the discussion regarding turbine first stage pressure in SE Section 3.11.7, the NRC staff concludes that the proposed changes to TS 3.3.4.2 are acceptable.

3.11.10 TS 3.4.2 – Jet Pumps

SR 3.4.2.1 includes a note that states that the SR is not required to be performed "until 24 hours after > 23% RTP." The proposed amendments would change the "23% RTP" value to "22.6% RTP."

As discussed in the TS Bases, the note allows this SR to not be performed when thermal power is below the currently specified 23 percent RTP limit (i.e., during low flow conditions). During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data.

As discussed in Section 2.2 of Attachment 1 to the application, these proposed changes are based on rescaling of the thermal limits monitoring threshold. The NRC staff's review of the scaling of the thermal limits monitoring threshold is discussed in SE Section 3.11.3.

Based on the discussion in SE Section 3.11.3, the NRC staff concludes that the proposed changes to TS 3.4.2 are acceptable.

3.11.11 TS 3.5.1 – ECCS – Operating and TS 3.5.3 – RCIC System

SR 3.5.1.8 provides requirements for verification of the HPCI pump flow rate. Similarly, SR 3.5.3.3 provides requirements for verification of the RCIC pump flow rate. Both SRs currently require the flow tests to be performed when reactor pressure is between 915 and 1,053 psig. The licensee has proposed to change the lower reactor pressure limit for both SRs from 915 psig to 910 psig.

As discussed in Section 3.4 of Attachment 1 to the application, the licensee stated, in part, that:

> As a result of the PBAPS Units 2 and 3 MUR LAR, the reactor pressure at rated MUR conditions will remain the same as CLTP conditions. To retain the reactor pressure the same at full power, the steam pressure at the main turbine stop valves must be lowered from 915 to approximately 910 psig to accommodate the increase in steam flow rate and the resulting higher steam pressure drop across

> the main steam lines and valves. The lower reactor pressure limit for performing the HPCI and RCIC system tests will also be changed from 915 to 910 psig to align with plant startup operations. This change will reduce challenges to the control of reactor pressure and reactivity when performing the surveillances during plant startup. A similar change was made following the EPU and was approved in Amendments 308 and 312, for Unit 2 and Unit 3, respectively [ADAMS Accession No. ML16159A148].
>
> An evaluation was performed at MUR conditions of the technical analysis of the similar change made for EPU conditions. The evaluation concluded the technical justification remains applicable to the change in the lower reactor pressure limit for performing the HPCI and RCIC system tests from 915 to 910 psig at MUR conditions. The current and proposed lower values of 915 psig and 910 psig, respectively, are consistent with the minimum EHC [electrohydraulic control] pressure setpoint at which reactor power can be increased without the need to adjust the EHC pressure setpoint during operation in Mode 1. Lowering the lower test pressure from 915 to 910 psig does not impact when the performance of the test is required. Neither the required HPCI and RCIC pump flow rates of 5000 gpm and 600 gpm, respectively, nor the pump discharge pressures are being changed.

The NRC staff reviewed the licensee's justifications for the proposed change as discussed above, as well as the discussion in Section 3.1 of the NRC staff SE for similar changes to SR 3.5.1.8 and SR 3.5.3.3 as part of PBAPS, Units 2 and 3, Amendment Nos. 308 and 312 (ADAMS Accession No. ML16159A148). The staff concludes that the proposed change is acceptable since the 910 psig value is consistent with the minimum EHC pressure setpoint at which reactor power can be increased without the need to adjust the EHC pressure setpoint during operation in Mode 1. In addition, the proposed change continues to meet the intent of 10 CFR 50.36(c)(3) with respect to establishing SRs that provide assurance that the associated LCOs will be met.

3.11.12 <u>TS 3.7.6 – Main Turbine Bypass System</u>

The TS 3.7.6 LCO Applicability and LCO Required Action B.1 currently include requirements associated with a thermal power limit of "23% RTP." The proposed amendments would change "23% RTP" value to "22.6% RTP" in both instances.

As discussed in the TS Bases, the main turbine bypass system is required to be operable above the specified thermal power level (i.e., currently "23% RTP") to ensure that the fuel cladding integrity safety limit and the cladding 1 percent plastic strain limit are not violated during the applicable safety analysis transients. As discussed in the Bases for LCO 3.2.3, "Linear Heat Generation Rate (LHGR)," and LCO 3.2.2, "Minimum Critical Power Ratio (MCPR)," sufficient margin to these limits exists below the specified thermal power level; therefore, these requirements are only necessary when operating at or above this power level.

As discussed in SE Sections 3.11.5 and 3.11.6, the proposed new thermal power limit applicable for the MCPR and LHGR LCOs (i.e., "22.6% RTP") is based on the scaling of the thermal limits monitoring threshold. Consistent with the TS Bases discussion above, the proposed changes to TS 3.7.6 are based on the scaling of the thermal limits monitoring threshold. The NRC staff's review of the scaling of the thermal limits monitoring threshold is discussed in SE Section 3.11.3.

Based on the discussion in SE Section 3.11.3, the NRC staff concludes that the proposed changes to TS 3.7.6 are acceptable.

3.11.13 TS Bases and Technical Requirements Manual

Attachment 3 to the application provided revised TS Bases pages to be implemented with the associated TS changes.

Attachment 3 to the application also provided revised pages to the Technical Requirements Manual (TRM) for information only. Nuclear Energy Institute (NEI) guidance document NEI 98-03, Revision 1, "Guidelines for Updating Final Safety Analysis Reports" (ADAMS Accession No. ML003779028), page 7 of Appendix A, lists the following methods of controlling the TRM:

> The TRM or other licensee controlled document is explicitly "incorporated by reference" into the UFSAR. Under this approach, the referenced document is subject to the change control requirements of 10 CFR 50.59 and the update/reporting requirements of 10 CFR 50.71(e), e.g., periodic submittal of change pages, etc.
>
> The TRM or other licensee controlled document is treated in a manner consistent with procedures fully or partially described in the UFSAR. Under this approach, the referenced document is maintained on-site in accordance with licensee administrative processes, and changes are evaluated using 10 CFR 50.59.

Regulatory Guide (RG) 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," dated September 1999 (ADAMS Accession No. ML992930009), states that Revision 1 of NEI 98-03 provides methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.71(e).

The TRM is described in Section 13.6.8 of the PBAPS UFSAR. This section of the UFSAR states, in part, that in accordance with NEI 98-03, Revision 1, the TRM is treated in a manner consistent with procedures fully or partially described in the UFSAR. As such, changes to the TRM are controlled under the provisions of 10 CFR 50.59.

3.12 Technical Evaluation Conclusion

Based on the considerations discussed in SE Sections 3.2 through 3.11, the NRC staff concludes that the proposed TPO uprate is acceptable.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments on October 16, 2017. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (May 2, 2017; 82 FR 20497). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

## 6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

## 7.0 REFERENCES

1. Exelon letter to NRC dated February 17, 2017, "Peach Bottom Atomic Power Station, Units 2 and 3, Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate" (ADAMS Accession No. ML17048A444).

2. Exelon letter to NRC dated March 20, 2017, "Peach Bottom Atomic Power Station, Units 2 and 3, Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate – Supplement 1 – Request for Non-Proprietary Version of Cameron Corporation Proprietary Documents" (ADAMS Accession No. ML17080A067).

3. Exelon letter to NRC dated July 13, 2017, "Peach Bottom Atomic Power Station, Units 2 and 3, Measurement Uncertainty Recapture License Amendment Request – Supplement 2 Response to Request for Additional Information" (ADAMS Accession No. ML17195A285).

4. NRC letter to Exelon dated August 25, 2014, "Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Extended Power Uprate (TAC Nos. ME9631 and ME9632)" (Amendment Nos. 293 and 296) (ADAMS Accession No. ML14133A046).

5. NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002 (ADAMS Accession No. ML013530183).

6. GE Nuclear Energy, Licensing Topical Report, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," NEDC-32938P-A, Revision 2, dated May 2003 (referred to as the TLTR) (ADAMS Accession Nos. ML17076A205 and ML17076A206, non-public) and ADAMS Package No. ML17076A207, public).

7. NRC letter to Exelon dated March 21, 2016, "Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Maximum Extended Load Line Limit Analysis Plus (CAC Nos. MF4760 and MF4761)" (Amendment Nos. 305 and 309) (ADAMS Accession No. ML16034A372).

8. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-33173P-A, Revision 4, "Applicability of GE Methods to Expanded Operating Domains," dated November 2012 (ADAMS Package Accession No. ML123130130).

9. 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 Package Accession No. ML091800530).

10. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-33075P-A, Revision 8, "GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," dated November 2013 (ADAMS Package Accession No. ML13324A393).

11. 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 Package Accession No. ML110970401).

12. Exelon letter to NRC dated August 8, 2017, "Peach Bottom Atomic Power Station, Units 2 and 3, Measurement Uncertainty Recapture License Amendment Request – Supplement 3 Response to Request for Additional Information" (ADAMS Accession No. ML17220A214).

13. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792).

14. NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2, dated May 2002 (ADAMS Package Accession No. ML021700373).

15. NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, dated November 2012 (ADAMS Accession No. ML12324A013).

16. NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, dated September 2007 (ADAMS Accession No. ML072640413).

17. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," dated March 2007 (ADAMS Package Accession No. ML070660036).

18. Electric Power Research Institute Topical Report No. 1025144NP, "BWRVIP-86NP, Revision 1-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," Final Report, dated May 2013 (ADAMS Accession No. ML13176A097).

19. Letter from NRC (R. Nelson) to Exelon (D. Czufin), "Final Safety Evaluation for Electric Power Research Institute Boiling Water Reactor Vessel and Internals Project Technical Report 1016575, 'BWRVIP-86, Revision 1: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan,'" dated October 20, 2011 (ADAMS Package Accession No. ML112780511).

20. NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, dated May 1988 (ADAMS Accession No. ML003740284).

21. NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," dated October 14, 2014 (ADAMS Accession No. ML14149A165).

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23. Letter from BWRVIP (C. Terry) to NRC (M. Khanna) "Project 704 – 'BWRVIP-74-A: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal," dated July 18, 2003 (ADAMS Package Accession No. ML031710354).

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26. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001 (ADAMS Accession No. ML010890301).

27. NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protections System Limits," dated January 31, 1996 (ADAMS Accession No. ML031110004).

28. GE-Hitachi Nuclear Energy, Licensing Topical Report NEDO-33178-A, "GE Hitachi Nuclear Energy, Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 1, dated June 2009 (ADAMS Accession No. ML092370487).

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31. NRC letter to Exelon dated January 24, 2012, "Peach Bottom Atomic Power Station, Units 2 and 3 – Requests for Relief I4R-51 and I4R-52 (TAC Nos. ME5392, ME5393, ME5394 and ME5395)" (ADAMS Accession No. ML112770217).

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Appendices:
A. List of Acronyms

## APPENDIX A
## LIST OF ACRONYMS

| ACRONYM | DEFINITION |
|---|---|
| AC | Alternating Current |
| ADAMS | Agencywide Documents Access and Management System |
| ADS | Automatic Depressurization System |
| AEC | Atomic Energy Commission |
| ALARA | As Low as is Reasonably Achievable |
| AOO | Anticipated Operational Occurrences |
| AOPs | Abnormal Operating Procedures |
| AOT | Allowed Outage Time |
| 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 |
| BOP | Balance of Plant |
| BSP | Backup Stability Protection |
| BTU | British Thermal Unit |
| BWR | Boiling-Water Reactor |
| BWRVIP | Boiling-Water Reactors Vessel and Internals Project |
| CF | Core Flow |
| CFD | Condensate Filter/Demineralizers |
| CFR | *Code of Federal Regulations* |
| CLTP | Current Licensed Thermal Power |
| CPR | Critical Power Ratio |
| CRD | Control Rod Drive |
| CW | Chilled Water |
| CS | Core Spray |
| DBA | Design-Basis Accident Loss-of-Coolant Accident |
| DBLOCA | Design-Basis |
| DC | Direct Current |
| DSS-CD | Detect and Suppress Solution - Confirmation Density |
| DVR | Degraded Voltage Relay |
| ECCS | Emergency Core Cooling System |
| ECT | Emergency Cooling Tower |
| EDG | Emergency Diesel Generator |
| EFPY | Effective Full Power Years |
| EHC | Electro-Hydraulic Control |
| EMA | Equivalent Margins Analysis |

| ACRONYM | DEFINITION |
|---|---|
| EOC | End-of-Cycle |
| EOPs | Emergency Operating Procedures |
| EPRI | Electric Power Research Institute |
| EPU | Extended Power Uprate |
| ES | Extraction Steam |
| ESF | Engineered Safety Features |
| ESW | Emergency Service Water |
| EQ | Environmental Qualification |
| °F | Degrees Fahrenheit |
| FAC | Flow-Accelerated Corrosion |
| FIV | Flow-Induced Vibration |
| FPCCS | Fuel Pool Cooling and Cleanup System |
| FSSD | Fire Safe-Shutdown |
| FW | Feedwater |
| GDC | General Design Criterion/Criteria |
| GE | General Electric |
| GEH | GE-Hitachi Nuclear Energy |
| GL | Generic Letter |
| GNF | Global Nuclear Fuel |
| gpm | gallons per minute |
| HCTL | Heat Capacity Temperature Limit |
| HELB | High Energy Line Break |
| HPCI | High-Pressure Coolant Injection |
| HPSW | High-Pressure Service Water |
| HVAC | Heating, Ventilating, and Air Conditioning |
| ICF | Increased Core Flow |
| IORV | Inadvertent Opening of a Relief Valve |
| ISI | Inservice Inspection |
| ISP | Integrated Surveillance Program |
| kV | Kilovolt |
| kW/ft | Kilowatts per Foot |
| LAR | License Amendment Request |
| LCO | Limiting Condition for Operation |
| LEFM | Leading Edge Flow Meter |
| LHGR | Linear Heat Generation Rate |
| LOCA | Loss-of-Coolant Accident |
| LOFW | Loss of Feedwater |
| LOOP | Loss of Offsite Power |
| LPCI | Low Pressure Coolant Injection |
| LPRM | Local Power Range Monitor |
| LR | License Renewal |
| LTR | Licensing Topical Report |
| MAPLHGR | Maximum Average Planar Linear Heat Generation Rate |

| ACRONYM | DEFINITION |
|---|---|
| MASR | Minimum Alternating Stress Ratio |
| MCO | Moisture Carry-Over |
| MCPR | Minimum Critical Power Ratio |
| M+ | (Short for MELLLA+) |
| MELLLA | Maximum Extended Load Line Limit Analysis |
| MELLLA+ | Maximum Extended Load Line Limit Analysis Plus |
| Mlbm/hr | Million Pounds Mass Per Hour |
| MOV | Motor-Operated Valve |
| MS | Main Steam |
| MSIV | Main Steam Isolation Valve |
| MSIVC | Main Steam Isolation Valve Closure |
| MSL | Main Steam Line |
| MSLB | Main Steam Line Break |
| MUR | Measurement Uncertainty Recapture |
| MVA | Megavolt Amp |
| MWd/ST | Megawatt-days per Short Ton |
| MWe | Megawatts Electric |
| MWt | Megawatts Thermal |
| NEI | Nuclear energy Institute |
| NPSH | Net Positive Suction Head |
| NPSHA | Net Positive Suction Head Available |
| NRC | U.S. Nuclear Regulatory Commission |
| NSSS | Nuclear Steam Supply System |
| OLMCPR | Operating Limit Minimum Critical Power Ratio |
| OLTP | Original Licensed Thermal Power |
| OPRM | Oscillation Power Range Monitor |
| PBAPS | Peach Bottom Atomic Power Station |
| PCLRT | Primary Containment Leak Rate Test |
| PCT | Peak Cladding Temperature |
| PECO | Philadelphia Electric Company |
| PFM | Probabilistic Fracture Mechanics |
| PMS | Plant Monitoring System |
| PRFO | Pressure Regulator Failure Open |
| psi | Pounds per Square Inch |
| psia | Pounds per Square Inch Atmospheric |
| psig | Pounds per Square Inch Gauge |
| P-T | Pressure-Temperature |
| PTLR | Pressure-Temperature Limits Report |
| RAI | Request for Additional Information |
| RBCCW | Reactor Building Closed Cooling Water |
| RBM | Rod Block Monitor |
| RCIC | Reactor Core Isolation Cooling |
| RCPB | Reactor Coolant Pressure Boundary |

| ACRONYM | DEFINITION |
|---|---|
| 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 |
| RIS | Regulatory Issue Summary |
| rpm | Revolutions per Minute |
| RPT | Recirculation Pump Trip |
| RPV | Reactor Pressure Vessel |
| RRS | Reactor Recirculation System |
| RTP | Rated Thermal Power |
| RWCU | Reactor Water Cleanup |
| RWM | Rod Worth Minimizer |
| SAFDL | Specified Acceptable Fuel Design Limit |
| SAR | Safety Analysis Report |
| SBO | Station Blackout |
| SE | Safety Evaluation |
| SLC | Standby Liquid Control |
| SLMCPR | Safety Limit Minimum Critical Power Ratio |
| SFP | Spent Fuel Pool |
| SLO | Single Loop Operation |
| SR | Surveillance Requirement |
| SRP | Standard Review Plan |
| SRV | Safety Relief Valve |
| SSCs | Structures, Systems, and Components |
| SW | Service Water |
| TAF | Top of Active Fuel |
| TCD | Thermal Conductivity Degradation |
| TIP | Traversing Incore Probes |
| TLO | Two Loop Operation |
| TLTR | TPO Licensing Topical Report |
| Tmin | Minimum Temperature for Stable Film Boiling |
| TPO | Thermal Power Optimization |
| TRM | Technical Requirements Manual |
| TS | Technical Specification |
| TSAR | Thermal Power Optimization Safety Analysis Report |
| TTWBP | Turbine Trip With Bypass |
| UFM | Ultrasonic Flow Meters |
| UFSAR | Updated Final Safety Analysis Report |
| USE | Upper Shelf Energy |
| wt. % | Weight Percentage |

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – ISSUANCE OF AMENDMENTS RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (CAC NOS. MF9289 AND MF9290; EPID L-2017-LLS-0001) DATED NOVEMBER 15, 2017

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RidsNrrDraAplb Resource
RidsNrrDraAphb Resource
RidsNrrDraArcb Resource
JTobin, NRR
RHaskell, NRR
JHughey, NRR
EDickson, NRR
CBasavaraju, NRR
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MRazzaque, NRR
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KHeller, NRR
SJones, NRR
MHamm, NRR
NIqbal, NRR
RidsNrrDmlr Resource
RidsNrrDmlrMccb Resource
RidsNrrDmlrMphb Resource
RidsNrrDmlrMvib Resource

**ADAMS Accession No.: Package: ML17286A014**
**Cover Letter and Enclosures 1, 2, and 3: ML17286A013**
**Enclosure 4 (Proprietary SE): ML17285A224** *by e-mail

| | | | | | |
|---|---|---|---|---|---|
| OFFICE | DORL/LPL1/PM | DORL/LPL1/LA | DE/EEOB/BC | DE/EICB/BC | DE/ESEB/BC |
| NAME | REnnis | LRonewicz | JQuichocho | MWaters | BWittick |
| DATE | 11/14//2017 | 10/18/2017 | 10/30/2017 | 10/20/2017 | 10/26/2017 |
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| NAME | DAlley (JTsao for) | SRuffin | KHsueh* | GCasto | SWeerakkody* |
| DATE | 10/30/2017 | 10/24/2017 | 10/30/2017 | 10/25/2017 | 10/30/2017 |
| OFFICE | DSS/SBPB/BC | DSS/SNPB/BC | DSS/SRXB/BC | DSS/STSB/BC(A) | DMLR/MCCB/BC |
| NAME | RDennig* | RLukes* | EOesterle | JWhitman | SBloom |
| DATE | 10/19/2017 | 10/30/2017 | 10/23/2017 | 10/20/2017 | 10/19/2017 |
| OFFICE | OGC | DORL/LPL1/BC | DORL/DD | DORL/LPL1/PM | |
| NAME | DRoth | JDanna | EBenner | REnnis | |
| DATE | 11/9/2017 | 11/15/2017 | 11/15/2017 | 11/15/2017 | |

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