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

April 1, 2013

Mr. Michael J. Pacilio
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: RELOCATION OF PRESSURE AND TEMPERATURE LIMIT CURVES TO THE PRESSURE AND TEMPERATURE LIMITS REPORT (TAC NOS. ME8535 AND ME8536)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendments Nos. 286 and 289 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 (TSs) and Facility Operating Licenses in response to your application dated April 27, 2012, as supplemented by letter dated October 15, 2012.

The amendments: (1) adopt a new methodology for preparation of the reactor coolant system pressure-temperature (P-T) limits, (2) relocate the P-T limits in the TSs to a new licensee-controlled document, the Pressure and Temperature Limits Report (PTLR), and (3) modify the TSs to add references to the PTLR.

A copy of the safety evaluation is also enclosed Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

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

Docket Nos 50-277 and 50-278

Enclosures:
1. Amendment No. 286 to Renewed DPR-44
2. Amendment No. 289 to Renewed DPR-56
3. Safety Evaluation

cc w/encls: 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. 286
Renewed License No. DPR-44

1. The 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 April 27, 2012, as supplemented by letter dated October 15, 2012, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 286, are hereby incorporated in the renewed 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications
and Facility Operating License

Date of Issuance: April 1, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 286

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 a marginal line indicating the area 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
1.1-6	1.1-6
3.4-21	3.4-21
3.4-22	3.4-22
3.4-23	3.4-23
3.4-24	3.4-24
3.4-25	3.4-25
3.4-26	3.4-26
3.4-27	3.4-27
5.0-22	5.0-22
- - -	5.0-22a

(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 3514 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 286, 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.

(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 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 3514 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. When using this definition to suspend the Applicability of LCOs, secondary containment ground-level hatches H15, H16, H17, H18, H19, and H33 shall be closed during the movement of any irradiated fuel in Secondary Containment.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and 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)

1.1 Definitions (continued)

STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components: a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE-------- Required Action A.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 AND A.2	Restore parameter(s) to within limits. Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 AND B.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify: a. RCS pressure and RCS temperature are within the limits specified in the PTLR; and b. RCS heatup and cooldown rates are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in the PLTR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	----------NOTE---------- Only required to be performed when tensioning the reactor vessel head bolting studs. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.9.6	----------NOTE---------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.9.7	----------NOTE---------- Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.

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5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis";

8. PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations"; and

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

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

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

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

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

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

(continued)

5.6 Reporting Requirements

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

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

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

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

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. 289
Renewed License No. DPR-56

1. The 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 April 27, 2012, as supplemented by letter dated October 15, 2012, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 289, are hereby incorporated in the renewed 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications
and Facility Operating License

Date of Issuance: April 1, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 289

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 a marginal line indicating the area 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
1.1-6	1.1-6
3.4-21	3.4-21
3.4-22	3.4-22
3.4-23	3.4-23
3.4-24	3.4-24
3.4-25	3.4-25
3.4-26	3.4-26
3.4-27	3.4-27
5.0-22	5.0-22
- - -	5.0-22a

(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 3514 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 289, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.[1]

(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[2], 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.

[1]Licensed power level was revised by Amendment No. 250, dated November 22, 2002, and will be implemented following the 14th refueling outage currently scheduled for Fall 2003.

[2]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 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 3514 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. When using this definition to suspend the Applicability of LCOs, secondary containment ground-level hatches H20, H21, H22, H23, H24, and H34 shall be closed during the movement of any irradiated fuel in Secondary Containment.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and 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)

1.1 Definitions (continued)

STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components: a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.9.1 ----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify: a. RCS pressure and RCS temperature are within the limits specified in the PTLR; and b. RCS heatup and cooldown rates are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. ---------- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	----------NOTE---------- Only required to be performed when tensioning the reactor vessel head bolting studs. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.9.6	----------NOTE---------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.9.7	----------NOTE---------- Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program.

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5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis";

8. PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations"; and

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

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

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

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

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

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

5.6 Reporting Requirements

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

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 286 AND 289

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

1.0 INTRODUCTION

By application dated April 27, 2012, as supplemented by letter dated October 15, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML121230354 and ML12290A113, respectively), Exelon Generation Company, LLC (Exelon, the licensee), requested changes to the Technical Specifications (TSs) and Facility Operating Licenses (FOLs) for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3.

The proposed amendment would: (1) adopt a new methodology for preparation of the reactor coolant system pressure-temperature (P-T) limits, (2) relocate the P-T limits in the TSs to a new licensee-controlled document, the Pressure and Temperature Limits Report (PTLR), and (3) modify the TSs to add references to the PTLR. PBAPS, Units 2 and 3, are currently licensed to P-T limits that are applicable up to 32 effective full-power years (EFPY). The PTLR would include P-T limits applicable to both 32 EFPY and 54 EFPY.

The supplement dated October 15, 2012, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 3, 2012 (77 FR 39525).

2.0 REGULATORY EVALUATION

The NRC has established requirements in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates the acceptability of a facility's proposed P-T limits based on the following NRC regulations and guidance:

(1) Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50,

(2) Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50,

(3) Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials,"

(4) Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity,"

(5) GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," and

(6) Standard Review Plan (SRP) Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock."

Appendix G to 10 CFR Part 50 requires that facility P-T limits for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code). Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation.

GL 92-01, Revision 1 requested that licensees submit the RPV data for their plants to the NRC staff for review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. SRP Section 5.3.2 provides an acceptable method for determining the P-T limits for ferritic materials in the beltline of the RPV based on the ASME Code Appendix G methodology.

The most recent version of Appendix G to Section XI of the ASME Code which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G, is the 2008 Edition of the ASME Code. This edition of Appendix G to Section XI of the ASME Code incorporates the provisions of ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," and ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves." Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20% of the preservice hydrostatic test pressure.

GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," provided guidance to licensees for development of a license amendment request to relocate P-T limit curves from the TSs to a PTLR.

Additional guidance regarding the proposed TS changes for relocation of P-T limits to a PTLR is provided in TS Task Force (TSTF) Traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS [Improved Standard Technical Specification] 5.6.6, RCS [Reactor Coolant System] PTLR," and in an NRC letter to the TSTF dated August 4, 2011 (ADAMS Accession No. ML110660285).

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The revised P-T limits for PBAPS, Units 2 and 3, are based on application of the methodology in GE-Hitachi Nuclear Energy (GEH) Licensing Topical Report NEDC-33178P-A, Revision 1, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves" (ADAMS Accession No. ML092370487). NEDC-33178P-A (henceforth the GEH methodology) provides the NRC-approved generic methodology, for General Electric-designed boiling-water reactors (BWRs), for generating P-T limits based on the plant-specific adjusted reference temperature (ART). The GEH methodology provides beltline and generic upper vessel and bottom head P-T limit curves that are shifted by the plant-specific ART, as well as guidance on the application of the ASME Code, Appendix G and 10 CFR Part 50, Appendix G.

For the RPV beltline material, the licensee identified plate C2873-1 as the limiting beltline material for PBAPS, Unit 2. For PBAPS, Unit 3, the limiting RPV beltline material was cited as being plate C2773-2. ART values were calculated for 32 and 54 EFPY. The licensee noted that the N16 water level instrument nozzle was evaluated using the adjoining shell ring #2 material at that location as the limiting material. The parameters used to determine the licensee's ART values for the limiting materials at the one-quarter of the RPV wall thickness (1/4T) location for 32 and 54 EFPY are shown in Appendices B and C of Attachment 5 to the application dated April 27, 2012. Corresponding parameters at the three-quarter of the RPV wall thickness (3/4T) were not provided in the attachments. Instead, the licensee applied the maximum tensile stress for both heatup and cooldown at the 1/4T location. The licensee stated that this approach is conservative as the 1/4T material toughness is lower than that in the 3/4T locations.

P-T limit Curves A, B, and C for both units are provided in Appendices B and C of Attachment 5 to the application, and are based on application of the GEH methodology. The licensee noted on page 9 of Attachment 5 that:

> the [PBAPS, Unit 2] P-T curves are not beltline limited for Curves A, B, or C, for 32 or 54 EFPY... the [PBAPS, Unit 3] P-T curves are not beltline limited for Curves A, B, or C for 32 EFPY. For 54 EFPY, Curve A is beltline limited at pressures above 1070 psig and Curves B and C are beltline limited at pressures above 1160 psig.

The licensee provided data from the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), BWRVIP-135, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations," consistent with a requirement in the GEH methodology. However, as the target materials did not match the representative materials, the data from BWRVIP-135 was not used or found to be limiting for PBAPS, Units 2 and 3. Information was also included detailing the determination process for evaluating non-beltline but possibly limiting components.

3.2 NRC Staff Evaluation

3.2.1 PTLR Methodology Implementation

The licensee utilized the GEH methodology to develop their PTLR. The GEH methodology was approved for use in generating PTLRs by the NRC staff. The NRC staff examined the proposed PBAPS, Units 2 and 3, PTLR (Attachment 5 to the application dated April 27, 2012) and determined that it was developed from the template PTLR found in the GEH methodology.

As discussed in Section 2.0 of this safety evaluation (SE), GL 96-03 provided guidance to licensees for development of a license amendment request to relocate P-T limit curves from the TSs to a PTLR. Attachment 1 to the GL contains seven technical criteria that must be met for the PTLR to be acceptable. The NRC staff reviewed the proposed PBAPS PTLR against the technical criteria discussed in GL 96-03 as follows:

(1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluences.

Section 3.0 of the proposed PBAPS PTLR documents that the neutron fluence was calculated per the NRC-approved methodology NEDC-32983P-A, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (ADAMS Accession No. ML072480121). This approved report documents the transport calculation methods including computer codes and formula used to calculate neutron fluences. Hence, the first criterion is met for PBAPS.

(2) The PTLR methodology describes the surveillance program.

Appendix A of the proposed PBAPS PTLR documents that PBAPS, Units 2 and 3, have participated in the approved BWRVIP Integrated Surveillance Program (BWRVIP-135), which meets the requirements of 10 CFR Part 50, Appendix H. Hence, the second criterion is met for PBAPS.

(3) The PTLR methodology describes how the low temperature overpressure protection system limits are calculated applying system/thermal hydraulics and fracture mechanics.

This criterion is not applicable to BWRs, and PBAPS, Units 2 and 3, are BWR units.

(4) The PTLR methodology describes the method for calculating the ART values using RG 1.99, Revision 2.

Section 5.0 of the proposed PBAPS PTLR indicated that RG 1.99, Revision 2 provided the methods for determining the ARTs for the PBAPS beltline materials, with their chemistry factors determined by surveillance data information from the BWRVIP ISP. Hence, the fourth criterion is met for PBAPS.

(5) The PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on ASME Code, Section XI, Appendix G, and the SRP.

Section 3.0 of the proposed PBAPS PTLR states that the P-T limits were calculated in accordance with the NRC-approved GEH methodology. This description is sufficient as the GEH methodology was reviewed against the criteria in GL 96-03 and found to satisfy the fifth criterion. Hence, the fifth criterion is met for PBAPS.

(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T limits for boltup temperature and hydrotest temperature.

Again, referencing the GEH methodology in the proposed PBAPS PTLR is sufficient because the methodology contains detailed information regarding the minimum temperature requirements for the boltup temperature and hydrotest temperature. The GEH methodology was reviewed against the criteria in GL 96-03 and found to satisfy the sixth criterion. Hence, the sixth criterion is met for PBAPS.

(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation.

Again, referencing the GEH methodology is sufficient because the methodology contains detailed information regarding this criterion in its Appendix I. The GEH methodology was reviewed against the criteria in GL 96-03 and found to satisfy the seventh criterion. Hence, the seventh criterion is met for PBAPS.

Based on the above, the NRC staff concludes that the proposed PBAPS PTLR was implemented based on an approved methodology and meets the applicable technical criteria in GL 96-03.

3.2.2 Fluence Calculations

Page 3 of Attachment 1 to the application dated April 27, 2012, states, in part, that:

> As documented in Section 4.0 of the Safety Evaluation Report for GE Hitachi Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, licensees who choose to implement NEDC-33178P-A, Revision 1 as their facility's PTLR methodology must address one plant-specific action item:
>
> > The licensee must identify the report used to calculate the neutron fluence and document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology.
>
> Accordingly, the PTLR incorporates a fluence calculated in accordance with the GE Licensing Topical Report NEDC-32983P-A, Revision 2, which has been approved by the USNRC (Reference 5), and is in compliance with Regulatory Guide 1.190 [Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence]. The latest information from the BWRVIP Integrated Surveillance Program that is applicable to PBAPS, Units 2 and 3 has been utilized.

The neutron fluence values were calculated in accordance with the NRC-approved method described in GE-NEDO-32983-A, Revision 2 (the NEDO designator refers to the open distribution version of the NEDC report (ADAMS Accession No. ML072480121)). The NRC staff's SE approving NEDO-32983-A provides the staff's evaluation concluding that plant-specific neutron fluence values calculated following this methodology would be adherent to the RG 1.190 guidance and hence acceptable. RG 1.190 provides guidance concerning the calculation of acceptable reactor RPV neutron fluence values. Since the fluence calculations were performed in accordance with an NRC-approved methodology and using the guidance in RG 1.190, the NRC staff finds the fluence calculations acceptable insofar as they support the requested PTLR implementation.

Additionally, the NRC staff notes that, after 31.06 and 31.96 EFPY of exposure for Units 2 and 3, respectively, the licensee has calculated the fluence assuming the core is operating at 120% of its original licensed thermal power level. For operation at the current licensed thermal power (CLTP) level, this assumption will increase the neutron flux at the RPV surface over the value associated with CLTP operation, and therefore results in a higher fluence value. Because a higher fluence value results in an over-estimation of neutron-irradiation-induced damage, the assumption is conservative, which the NRC staff finds acceptable.

3.2.3 P-T Limits

The proposed P-T limits are a composite of the RPV beltline, the bottom head, and the upper vessel curves. Independent P-T curves generated by the NRC staff are consistent with P-T curves provided by the licensee. These curves were generated using the GEH methodology and ASME Code, Section XI, Appendix G.

To evaluate the proposed PBAPS, Units 2 and 3, RPV beltline P-T limits, the NRC staff first confirmed the licensee's selection of limiting materials. For the PBAPS, Units 2 and 3, beltline materials, the staff found that the initial RT_{NDT}, copper (Cu), and nickel (Ni) values are in agreement with the information in the NRC's Reactor Vessel Integrity Database (RVID). The licensee reported best estimate chemistry and ISP data from BWRVIP-86, Revision 1, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan" (ADAMS Accession No. ML090300555), to ensure the collection of credible chemistry and surveillance data. Best estimate chemistries from BWRVIP-86 do not significantly differ from the RVID, and therefore the inclusion of best estimate chemistry does not change the limiting beltline material previously identified by the staff. The licensee only calculated the ART values for the RPV 1/4T location. The staff concurs that this is appropriate as the licensee's approach of using the maximum tensile stress for either heatup or cooldown and applying it at the 1/4T location is equivalent to using the maximum thermal stress intensity factor (K_{IT}) and the minimum fracture toughness (K_{Ic}) in the heatup and cooldown analysis, making the proposed P-T limits bound both the heatup and cooldown curves.

As previously noted, the licensee made use of the GEH methodology in generating the P-T limits, with composite and limiting P-T limit Curves A, B, and C provided by the licensee. Composite curves reported by the licensee are consistent with composite curves generated by the NRC staff applying the GEH methodology, shifting the approved generic GE bottom curves by the ART for the limiting material identified. For Curve C below 20% of the hydro test

pressure (312 psig), the staff found the upper vessel curve generated using the GEH methodology limiting, consistent with the composite P-T curve provided by the licensee. For all other conditions, the Appendix G to 10 CFR Part 50 requirements for the minimum metal temperature of the closure head flange and vessel flange regions produce limiting "notches," serving to explain the distinct vertical lines at constant temperature above approximately 312 psig in the licensee's proposed P-T limits. For all PBAPS, Units 2 and 3, curves, a minimum temperature of 68 °F for the bottom head and 70 °F for the flange region was verified as being ASME Code compliant per the stipulation that these regions must be at least RT_{NDT} + 60 °F (where RT_{NDT} represents that property of the limiting material in the relevant region). When P > 312 psig, the minimum temperature of 100 °F for the pressure test curve, 130 °F for the normal operation/core not critical curve, and 170 °F for the normal operation/core critical curve are derived from adding the RT_{NDT} of 10 °F for the limiting flange material temperature to 90 °F, 120 °F, and 160 °F that were specified in Appendix G to 10 CFR Part 50 for the three operation conditions. The staff has also verified that when P ≤ 312 psig, the minimum temperatures of 68 °F (bottom head) and 70 °F (flange region) for the pressure test curve and the normal operation/core not critical curve is more conservative than the RT_{NDT} for the limiting flange material temperature that was specified in 10 CFR Part 50, Appendix G.

The licensee noted that nozzle N12, a beltline water level instrument nozzle, was evaluated. The NRC staff evaluated the disposition of this nozzle and other relevant nozzles and discontinuities and determined that they were adequately addressed in the implementation of the PTLR.

The NRC staff also reviewed the licensee's analysis of non-beltline components and materials. The licensee documented its evaluation of this in Attachment 4 of the application dated April 27, 2012. In many plant designs, the material properties of the beltline have been controlled such that geometric and non-beltline materials may in fact be the limiting factors in portions of the P-T limits. The staff requested that the licensee clarify further how the P-T limit curves in the submittal bounded all RPV materials and the lowest permissible service temperatures of all ferritic reactor coolant pressure boundary (RCPB) materials. In the supplement dated October 15, 2012, the licensee responded to the staff's request and confirmed that the P-T curves were developed to represent all vessel non-beltline discontinuities and provided details concerning this. The supplement dated October 15, 2012, also provided clarification regarding how certain NRC General Design Criteria (GDC), applicable to the PBAPS licensing basis were satisfied. As discussed in Appendix H of the PBAPS Updated Final Safety Analysis Report, PBAPS conforms to the intent of the GDC published by the Atomic Energy Commission (AEC) in the *Federal Register* for comment on July 11, 1967 (32 FR 10213). These GDC are typically referred to as the "draft GDC" since they pre-date the "final" GDC subsequently published by the AEC in the *Federal Register* on February 20, 1971 (36 FR 3255), and incorporated as Appendix A to 10 CFR Part 50. The licensee's supplement dated October 15, 2012, provided the following discussion regarding how draft GDC 35, "Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A)," was satisfied:

> Appropriate consideration is given in the design [of the RCPB] to the mechanical properties to ensure that, at the service temperatures, there is:

1. Complete energy absorption with fully ductile behavior (e.g., in the energy absorption region of 100 percent shear fracture) whenever the boundary can be pressurized beyond the systems safety valve setting by operational transients in postulated accidents.

2. An NDT temperature at least 60°F below the service temperature whenever the boundary can be pressurized beyond 20 percent of its design pressure by operational transients, hydrotests, and postulated accidents.

The above design approach is consistent with the Construction Code for PBAPS, Units 2 and 3; therefore, the NRC staff determined that the RCPB materials were adequately controlled with respect to the relevant engineering standards. The staff therefore finds the analysis of non-beltline RPV components and ferritic RCPB materials acceptable.

Based on the above evaluation, the NRC staff determined that the licensee's proposed P-T limits are in accordance with the NEDC-33178-A report and satisfy the requirements of Appendix G to Section XI of the ASME Code and Appendix G to 10 CFR Part 50. Hence, the licensee's proposed P-T limit curves are acceptable for operation of the PBAPS, Units 2 and 3, RPVs valid for 32 EFPY and 54 EFPY.

3.2.4 TS Changes

The proposed amendment would revise the PBAPS, Units 2 and 3, TSs as follows:

(1) TS Section 1.1, "Definitions," would add a new definition titled, "PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."

(2) TS Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits," would be revised to delete the P-T limit curves. In addition, reference to the curves would be replaced with reference to the PTLR.

(3) TS Section 5.6, "Reporting Requirements," would add a new Section 5.6.7 titled, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)." This section: (1) identifies the TSs that address the P-T limits (i.e., TS 3.4.9), (2) references the GEH methodology (including the specific revision and date) used to determine the P-T limits, and (3) requires that the PTLR be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

The NRC staff has reviewed the proposed TS changes and finds that they are consistent with the guidance provided in GL 96-03, as supplemented by TSTF-419-A and the guidance contained in the NRC's letter to the TSTF dated August 4, 2011 (ADAMS Accession No. ML110660285).

The licensee's application dated April 27, 2012, provided revised TS Bases pages to be implemented with the associated TS changes. These pages were provided for information only and will be revised in accordance with the TS Bases Control Program.

3.3 Technical Evaluation Summary and Conclusion

The NRC staff conclusions, based on the discussion in SE Sections 3.2.1 through 3.2.4, are summarized as follows:

(1) The NRC staff concludes that the proposed PBAPS PTLR was developed based on an approved methodology and meets the applicable technical criteria in GL 96-03.

(2) Since the fluence calculations were performed in accordance with an NRC-approved methodology and using the guidance in RG 1.190, the NRC staff finds the fluence calculations acceptable insofar as they support the requested PTLR implementation.

(3) The NRC staff determined that the licensee's proposed P-T limits are in accordance with the NEDC-33178-A report and satisfy the requirements of Appendix G to Section XI of the ASME Code and Appendix G to 10 CFR Part 50. Hence, the licensee's proposed P-T limit curves are acceptable for operation of the PBAPS, Units 2 and 3, RPVs valid for 32 EFPY and 54 EFPY.

(4) The NRC staff has reviewed the proposed TS changes and finds that they are consistent with the guidance provided in GL 96-03, as supplemented by TSTF-419-A and the guidance contained in the NRC's letter to the TSTF dated August 4, 2011 (ADAMS Accession No. ML110660285).

Based on the above, the NRC staff concludes that the proposed amendment 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. 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 surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (77 FR 39525). 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.

Principal Contributors: D. Widrevitz
B. Parks
R. Ennis

Date: April 1, 2013

April 1, 2013

Mr. Michael J. Pacilio
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: RELOCATION OF PRESSURE AND TEMPERATURE LIMIT CURVES TO THE PRESSURE AND TEMPERATURE LIMITS REPORT (TAC NOS. ME8535 AND ME8536)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendments Nos. 286 and 289 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 (TSs) and Facility Operating Licenses in response to your application dated April 27, 2012, as supplemented by letter dated October 15, 2012.

The amendments: (1) adopt a new methodology for preparation of the reactor coolant system pressure-temperature (P-T) limits, (2) relocate the P-T limits in the TSs to a new licensee-controlled document, the Pressure and Temperature Limits Report (PTLR), and (3) modify the TSs to add references to the PTLR.

A copy of the safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

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

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 286 to Renewed DPR-44
2. Amendment No. 289 to Renewed DPR-56
3. Safety Evaluation

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ADAMS Accession No: ML13079A219 *via email

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