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

July 28, 2015

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: ELIMINATE MAIN STEAM LINE RADIATION MONITOR TRIP AND ISOLATION FUNCTION (TAC NOS. MF4757 AND MF4758)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment Nos. 299 and 302 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the 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 September 3, 2014, as supplemented by letter dated February 9, 2015.

The amendments revise the TSs to eliminate the main steam line (MSL) radiation monitor from initiating: (1) a reactor protection system automatic reactor scram; and (2) a primary containment isolation system isolation, including automatic closure of the MSL isolation valves, MSL drain valves, MSL sample line valves, residual heat removal system sample line valves, and reactor recirculation loop sample line valves.

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. 299 to Renewed DPR-44
2. Amendment No. 302 to Renewed DPR-56
3. Safety Evaluation

cc w/enclosures: 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. 299
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 September 3, 2014, as supplemented by letter dated February 9, 2015, 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. 299, 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 prior to startup from the fall 2016 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical Specifications
and Facility Operating License

Date of Issuance: July 28, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 299

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
3.3-5	3.3-5
3.3-6a	3.3-6a
3.3-8	3.3-8
3.3-48	3.3-48
3.3-51	3.3-51
3.3-51a	3.3-51a
3.3-52	3.3-52

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

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

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 299, 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.

(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.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.10	Deleted.	
SR 3.3.1.1.11	------------------NOTES------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.16	Deleted.	
SR 3.3.1.1.17	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.18	Verify the RPS RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.19	Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 26.2% and recirculation drive flow is <60%.	In accordance with the Surveillance Frequency Control Program.

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 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 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 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 1.0 inches
5.	Main Steam Isolation Valve —Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
6.	Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 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 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 50.0 gallons
		5[a]	2	H	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 50.0 gallons
8.	Turbine Stop Valve—Closure	≥ 26.7% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
9.	Turbine Control Valve Fast Closure, Trip Oil Pressure—Low	≥ 26.7% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 500.0 psig
10.	Turbine Condenser—Low Vacuum	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 23.0 inches Hg vacuum
11.	Deleted					
12.	Reactor Mode Switch — Shutdown Position	1,2	1	G	SR 3.3.1.1.14 SR 3.3.1.1.17	NA
		5[a]	1	H	SR 3.3.1.1.14 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

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.a, 2.b, 8.a, and 8.b <u>AND</u> 24 hours for Functions other than Functions 2.a, 2.b, 8.a, and 8.b
B. One or more Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately

(continued)

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation 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 primary containment isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.6.1.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.3	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.4	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.6.1.6	Deleted.	
SR 3.3.6.1.7	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level — Low Low Low (Level 1)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ -160.0 inches
b. Main Steam Line Pressure — Low	1	2	E	SR 3.3.6.1.3 SR 3.3.6.1.7	≥ 825.0 psig
c. Main Steam Line Flow — High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 173.8 psid
d. Deleted					
e. Turbine Building Main Steam Tunnel Temperature — High	1,2,3	6	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 200.0°F
f. Reactor Building Main Steam Tunnel Temperature-High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 230.0°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level — Low (Level 3)	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
b. Drywell Pressure — High	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 2.0 psig
c. Main Stack Monitor Radiation — High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 2 X 10^{-2} µCi/cc
d. Reactor Building Ventilation Exhaust Radiation — High	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 16.0 mR/hr
e. Refueling Floor Ventilation Exhaust Radiation — High	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 16.0 mR/hr

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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. 302
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 September 3, 2014, as supplemented by letter dated February 9, 2015, 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. 302, 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 prior to startup from the fall 2015 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical Specifications
and Facility Operating License

Date of Issuance: July 28, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 302

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
3.3-5	3.3-5
3.3-6a	3.3-6a
3.3-8	3.3-8
3.3-48	3.3-48
3.3-51	3.3-51
3.3-51a	3.3-51a
3.3-52	3.3-52

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

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

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 302, 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.

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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.10	Deleted.	
SR 3.3.1.1.11	-------------------NOTES------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.16	Deleted.	
SR 3.3.1.1.17	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.18	Verify the RPS RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.19	Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥26.2% and recirculation drive flow is <60%.	In accordance with the Surveillance Frequency Control Program.

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 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 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 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 1.0 inches
5.	Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
6.	Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 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 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 50.0 gallons
		5(a)	2	H	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 50.0 gallons
8.	Turbine Stop Valve – Closure	≥ 26.7% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
9.	Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	≥ 26.7% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 500.0 psig
10.	Turbine Condenser – Low Vacuum	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 23.0 inches Hg vacuum
11.	Deleted					
12.	Reactor Mode Switch – Shutdown Position	1,2	1	G	SR 3.3.1.1.14 SR 3.3.1.1.17	NA
		5(a)	1	H	SR 3.3.1.1.14 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

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.a, 2.b, 8.a, and 8.b <u>AND</u> 24 hours for Functions other than Functions 2.a, 2.b, 8.a, and 8.b
B. One or more Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately

(continued)

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------
1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation 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 primary containment isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.6.1.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.6.1.3	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.6.1.4	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.6.1.5 Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.6.1.6 Deleted.	
SR 3.3.6.1.7 Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level — Low Low Low (Level 1)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ -160.0 inches
b. Main Steam Line Pressure — Low	1	2	E	SR 3.3.6.1.3 SR 3.3.6.1.7	≥ 825.0 psig
c. Main Steam Line Flow — High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 173.8 psid
d. Deleted					
e. Main Steam Tunnel Temperature — High	1,2,3	8	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 200.0°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level — Low (Level 3)	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
b. Drywell Pressure — High	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 2.0 psig
c. Main Stack Monitor Radiation — High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 2×10^{-2} μCi/cc
d. Reactor Building Ventilation Exhaust Radiation — High	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 16.0 mR/hr
e. Refueling Floor Ventilation Exhaust Radiation — High	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 16.0 mR/hr

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 299 AND 302

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 September 3, 2014, as supplemented by letter dated February 9, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14247A522 and ML15041A351, respectively), Exelon Generation Company, LLC (Exelon, the licensee), requested changes to the Technical Specifications (TSs) for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed amendments would revise the TSs to eliminate the main steam line radiation monitor (MSLRM) from initiating: (1) a reactor protection system (RPS) automatic reactor scram; and (2) a primary containment isolation system (PCIS) isolation, including automatic closure of the main steam line Isolation valves (MSIVs), main steam line (MSL) drain valves, MSL sample line valves, residual heat removal (RHR) system sample line valves, and reactor recirculation loop sample line valves. Existing requirements for the mechanical vacuum pump (MVP) would be retained in the Technical Requirements Manual (TRM). The licensee stated that elimination of these reactor trip and isolation functions of the MSLs improves the availability of the main condenser for removal of decay heat and aids in eliminating inadvertent reactor trips.

The licensee referenced Nuclear Regulatory Commission (NRC or the Commission) approved General Electric (GE) Licensing Topical Report (LTR) NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steamline Isolation Valve Closure Function and Scram Function of the Main Steamline Radiation Monitor" dated October 31, 1992 (ADAMS Legacy Library Accession No. 9604160114), as justification for eliminating the MSLRM trip and isolation functions from initiating a reactor scram and automatic closure of the MSIVs.

Enclosure 3

NEDO-31400A does not address the proposed changes to eliminate the automatic closure of the MSL drain valves, MSL sample line valves, RHR system sample line valves, and reactor recirculation loop sample line valves. The licensee provided additional information in the license amendment request (LAR) to justify these changes.

The supplement dated February 9, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 30, 2014 (79 FR 67201).

2.0 REGULATORY EVALUATION

2.1 Background

As discussed in Sections 1.6.2.20 and 7.12.1 of the PBAPS Updated Final Safety Analysis Report (UFSAR), the MSLRM consists of four gamma-sensitive radiation monitors located external to the four MSLs just outside of primary containment. The detectors are geometrically arranged so that the system is capable of detecting significant increases in gamma radiation levels. The safety objective of the MSLRM system is to monitor for the gross release of fission products from the fuel and, upon indication of such failure, to initiate appropriate action to limit fuel damage and contain the released fission products.

Under the current design, when a significant increase in MSL radiation is detected, the MSLRM system sends trip signals to the RPS and the PCIS. Upon receipt of the high radiation trip signals, the RPS initiates a scram; and the PCIS initiates closure of all MSIVs, MSL drain valves, MSL sample line valves, RHR system sample line valves, and reactor recirculation loop sample line valves. The MSLRM trip signal will also turn off the MVP (if it is running) and close the MVP suction valve.

Section 14.9.2.4 of the UFSAR discusses the current PBAPS design and licensing basis for a control rod drop accident (CRDA). This design-basis accident (DBA) currently credits the MSLRM system for closure of the MSIVs and stopping of the MVP upon detection of high radiation levels during a CRDA. The current CRDA also assumes that the steam jet air ejectors (SJAEs) are shut down upon detection of high radiation by the MSLRM system. The proposed amendments would revise the design and licensing basis for a CRDA such that upon detection of high radiation levels during a CRDA by the MSLRM, no credit is taken for MSIV closure or SJAE shutdown. Credit would still be taken for MVP cessation.

The MSLRM high radiation trip setpoint is selected sufficiently high enough above the rated full-power background radiation level in the vicinity of the MSLs so that spurious trips are avoided at rated power. The setpoint is low enough that the system can respond to the fission products released during the design-basis CRDA.

As discussed in the LAR, operating data presented in NEDO-31400A indicates that MSLRMs have initiated eight reactor shutdowns from 1980 through October 1992, but none of the shutdowns were the result of fuel degradation. The shutdowns were the result of instrument failures, chemistry excursions, radiation monitor maintenance errors, and other causes. The

proposed amendments would reduce the potential for unnecessary plant shutdowns and PCIS isolations caused by spurious MSLRM trips.

The licensee stated in the LAR that the justification for eliminating the MSLRM high radiation trip and isolation function from initiating an automatic reactor scram and automatic closure of the MSIVs is based on NEDO-31400A and the applicability of this report to PBAPS, Units 2 and 3. NEDO-31400A provides a safety assessment that demonstrates the MSIV isolation and scram functions of the MSLRMs are not required to ensure compliance with the accident dose guidelines of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100.

By letter dated May 15, 1991,[1] the NRC staff approved GE LTR NEDO-31400 and indicated that it would be acceptable for licensees to reference this report when submitting TS changes concerning the elimination of the MSLRM high radiation trip functions under certain conditions. In the NRC staff's safety evaluation (SE) enclosed with the letter dated May 15, 1991, the staff stated that participating boiling water reactor utilities listed in Table 1 of the LTR may reference NEDO-31400 in their LARs, provided that the following conditions are met (Note: PBAPS, Units 2 and 3, were listed in Table 1 of NEDO-31400):

1) The applicant demonstrates the assumptions with regard to input values (including power per assembly, χ/Q, and decay times) made in the generic analysis bound those for the plant;

2) The applicant includes sufficient evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the MSLs will be controlled expeditiously to limit both occupational doses and environmental releases; and,

3) The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint at 1.5 times the normal Nitrogen-16 background dose rate at the monitor locations, and commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if the MSLRM or offgas radiation monitors, or both, exceed their alarm setpoints.

2.2 Proposed TS Changes

The proposed amendments would remove requirements for the MSLRM trip and isolation functions from TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation," and Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation." The specific changes are as follows:

1) TS Table 3.3.1.1-1, Function 11, "Main Steam Line - High Radiation," would be deleted.

2) Surveillance Requirements (SRs) 3.3.1.1.10 and 3.3.1.1.16, which are associated with TS Table 3.3.1.1-1, Function 11, would be deleted.

3) TS Table 3.3.6.1-1, Function 1.d, "Main Steam Line - High Radiation," would be deleted.

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[1] The NRC letter dated May 15, 1991, is included as part of the approved version of the GE LTR (i.e., NEDO-31400A) dated October 31, 1992 (ADAMS Legacy Library Accession No. 9604160114).

4) Limiting Condition for Operation (LCO) 3.3.6.1, Condition A, would be revised to delete reference to TS Table 3.3.6.1-1, Function 1.d.

5) SR 3.3.6.1.3 would be revised to delete reference to TS Table 3.3.6.1-1, Function 1.d.

As noted in SE Section 1.0, existing requirements for the MVP would be retained in the TRM.

2.3 Regulatory Requirements and Guidance

General Design Criteria

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

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

- 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," which requires that core protection systems, together with associated equipment, be designed to act automatically to prevent or 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 engineered safety features.

- 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.

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 five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

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

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

Radiological Consequences

The regulatory requirements and guidance which the NRC staff considered in its review of this LAR with respect to radiological consequences included the following:

- 10 CFR 50.67, "Accident source term," which sets limits for the radiological consequences of a postulated 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).

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792), which provides guidance to licensees of operating power reactors on acceptable applications of ASTs.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports" (hereinafter referred to as SRP), Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000 (ADAMS Accession No. ML003734190), which, in part, provides guidance to NRC staff in performing reviews associated with the requirements in 10 CFR 50.67.

The dose acceptance criteria that the NRC staff used in its review of this LAR are based on the reference values in 10 CFR 50.67, the accident-specific acceptance criteria in Regulatory Position 4.4 of RG 1.183, and Table 1 of SRP 15.0.1. Specifically, the dose acceptance criteria for the CRDA are a total effective dose equivalent (TEDE) of 6.3 rem at the exclusion area boundary (EAB) for the maximum 2-hour period, 6.3 rem at the outer boundary of the low-population zone (LPZ) during the entire period of the postulated cloud passage, and 5 rem in the control room for the duration of the accident.

Human Factors

The guidance which the NRC staff considered in its review of this LAR with respect to human factors included the following:

- Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," dated October 23, 1997 (ADAMS Legacy Library Accession No. 9710230271), which provides guidance related to evaluating the acceptability of substituting manual actions for automatic actions.

- NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, dated September 30, 2007 (ADAMS Accession No. ML072640413), which provides guidance related to reviewing changes to operator actions credited in nuclear power plant safety analyses.

- NUREG-0800, SRP Chapter 18, "Human Factors Engineering," Revision 2, dated March 27, 2007 (ADAMS Accession No. ML070670253), which, in part, provides guidance to NRC staff in reviewing human factors considerations for LARs.

3.0 TECHNICAL EVALUATION

3.1 Human Factors

The following provides the NRC staff's evaluation of the LAR with respect to human factors considerations.

As discussed in the supplement dated February 9, 2015, the licensee identified the following two new operator actions that would be necessary as a result of the proposed amendments:

- Operators must isolate the reactor water (RW) sample lines (RHR and reactor recirculation loop sample lines) within 40 minutes of a CRDA, concurrent with the RW sample lines open. The operators must close the primary containment isolation valves (PCIVs) from the control room because the local controls are assumed to be unavailable due to radiation levels

exceeding allowable levels. Local alarms and personal dosimetry will alert operators of the need to evacuate the area around RW sample lines.

- Since the automatic MSIV trip will no longer be available, operators will need to isolate the MSIVs from the control room within 24 hours of a CRDA. Plant procedures will direct operators to initiate a manual shutdown in the case that adverse conditions continue to persist.

The licensee will use both of the actions described above to manually complete functions that were previously completed automatically; therefore, these actions were assessed by the NRC staff using Information Notice (IN) 97-78. The operator actions described above were also assessed in accordance with the generic risk categories established in Appendix A to NUREG-1764. This assessment was subsequently reviewed by an NRC risk analyst. The operator actions under review are not included in the operator actions listed on Table A.1, "Generic BWR Human Actions That Are Risk-Important," or Table A.2, "BWR Potentially Risk-Important Human Actions." These manual actions were, therefore, found to be appropriate for Level III, the lowest of the graded reviews possible under the guidance in NUREG-1764. However, since a Level III review is relatively superficial compared to the more conservative review process conducted using IN 97-78, IN 97-78 was used in lieu of the Level III review.

Information Notice 97-78 discusses items to consider when manual actions will be used to replace automatic actions such as: (1) the specific operator actions required; (2) the potentially harsh or inhospitable environmental conditions expected; (3) a general discussion of the ingress/egress paths taken by the operators to accomplish functions; (4) the procedural guidance for required actions; (5) the specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions; (6) any additional support personnel and/or equipment required by the operator to carry out actions; (7) a description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken; (8) the ability to recover from credible errors in performance of manual actions and the expected time required to make such a recovery; and (9) consideration of the risk significance of the proposed operator actions. The licensee addressed the applicable considerations as described below.

The application dated September 3, 2014, indicates that the alarm system, cues provided to operators, and control systems, will not change as a result of the proposed amendments. Although the automatic control functions will be removed as a result of these amendments, the associated alarm functions will remain, providing operators with indications of high radiation levels.

The supplement dated February 9, 2015, indicates that there are two new risk-important operator actions as described above. Manual completion of these tasks is relatively simple and will be driven by procedures. Changes to alarm response cards and transient response procedures will be evaluated by the licensee using the Engineering Change Request (ECR) process. The licensee also indicated that the existing ECR process will be used to assess the need for additional training and changes to the simulator. The licensee's Operations Training Department will evaluate changes in accordance with plant procedures. Changes to procedures/training will be communicated with operations personnel. Actions at local control

stations are not credited; the manual actions under review are all completed from the control room. Therefore, consideration of ingress/egress paths and harsh/inhospitable environments are not necessary for this analysis. Completion of these actions will not require additional personnel or equipment.

The supplement dated February 9, 2015, also describes the credible operator actions that are linked to the two operator manual actions. Identification and recovery of these errors is possible using the standard human performance verification tools.

The NRC staff finds that the new operator manual actions are relatively simple to accomplish, will be conducted in a safe environment, and ample time is available to complete the tasks, including time for recovery in the case of operator errors. Human performance verification tools, procedures, and training, are used to prevent errors from occurring. Based on the information provided in the application dated September 3, 2014, and the supplement dated February 9, 2015, as discussed above, the NRC staff concludes that the licensee has adequately addressed the considerations discussed in Information Notice 97-78. Therefore, the NRC staff concludes that the new operator manual actions are acceptable.

3.2 Reactor Systems

The following provides the NRC staff's evaluation of the LAR with respect to reactor systems.

The MSLRM system was designed to provide an early indication of gross fuel failures. The original intent of this system was to mitigate the releases of the detected fuel failures by providing a scram signal to terminate the initiating event and an MSIV closure signal to assure containment of the release.

Exelon has referenced NEDO-31400A in support of its request to eliminate the MSLRM scram and group isolation functions. In the topical report, GE analyzes a CRDA where the MSL high radiation isolation is eliminated. NEDO-31400A states that there have been a number of spurious actuations of the MSLRM system at plants causing unnecessary automatic reactor shutdowns (i.e., actuations were not due to failed fuel). In the supplement dated February 9, 2015, Exelon stated that one event involving a plant shutdown due to an MSLRM system initiation was identified for PBAPS, Unit 2. This event occurred on June 10, 1974, and was reported to the NRC on June 17, 1974. This event was the result of contaminants in the reactor coolant system which led to a high radiation signal (not due to a fuel failure) and automatic plant shutdown. The PBAPS operating experience validates that removing the MSLRM system high radiation reactor scram and MSIV closure functions will reduce the potential for unnecessary reactor shutdowns and will increase plant operational flexibility, since the main condenser will remain available for decay heat removal.

The proposed TS changes defeat portions of MSLRM high radiation trip function logic circuitry in the RPS and PCIS. In the supplement dated February 9, 2015, Exelon stated that although the PCIS system will no longer close the MSIVs; MSIV drain valves; or RHR, MSL, and reactor recirculation sample drain valves when an MSL high radiation signal is received, all other PCIS trip logic will remain unaffected by this change and will function as designed to perform their intended safety functions.

Exelon further stated that the amount of damaged fuel as a result of the CRDA is determined by a bounding analysis for a particular fuel design by the fuel vendor and is not impacted by this proposed LAR. As such, the number of fuel rods predicted to fail and melt is not changed.

The MSLRM system high radiation main control room (MCR) alarms and trip function for isolating the MVP will be retained. This will ensure that any radioactive material released from a fuel failure will be contained in the main condenser and processed through the offgas system, which continuously removes non-condensable gases from the main condenser by the SJAEs during plant operation. The offgas system reduces offgas radioactivity levels to permissible levels for release under all site atmospheric conditions. The system uses catalytic recombination for volume reduction and control of hydrogen concentration and activated charcoal filters to adsorb fission product and activation gases prior to release to the environment. Instrumentation permits system operation and monitoring from the MCR. All of the postulated radioactive material is assumed to be released to the condenser and turbine before the isolation occurs.

The NRC staff reviewed the application dated September 3, 2014; the supplement dated February 9, 2015; and related documentation (e.g., UFSAR, TSs). The staff concludes that the LAR is acceptable with respect to the impact on reactor systems based on the following considerations:

- The amount of damaged fuel as a result of the CRDA is determined by a bounding analysis for a particular fuel design and is not impacted by this LAR. As such, the number of fuel rods predicted to fail and melt is not changed as a result of the LAR.

- Although the proposed LAR would defeat portions of MSLRM high radiation trip function logic circuitry in the RPS and PCIS so that the PCIS system will no longer close the MSIVs; MSIV drain valves; or RHR, MSL, and reactor recirculation sample drain valves when an MSL high radiation signal is received; there are no impacts on the operation of the RPS or PCIS with respect to other intended safety functions.

- Removing the MSLRM system high radiation reactor scram and MSIV closure functions will increase plant operational flexibility since the main condenser will remain available for decay heat removal.

3.3 Radiological Consequences

The following provides the NRC staff's evaluation of the LAR with respect to radiological consequences.

Accident Source Term

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident source term," which provided a mechanism for licensed power reactors to voluntarily replace the traditional accident source term used in their DBA analyses with ASTs. Regulatory guidance for the implementation of these ASTs is provided in RG 1.183. Under 10 CFR 50.67, a licensee seeking to use the AST is required to apply for a license amendment, and the application is required to contain an evaluation of the consequences of DBAs.

The NRC approved a full scope implementation of an AST methodology for PBAPS by License Amendment Nos. 269 (Unit 2) and 273 (Unit 3) on September 5, 2008 (ADAMS Accession No. ML082320406). In the PBAPS AST amendments, the CRDA was based on the rapid removal (i.e., drop) of a highest worth control rod, resulting in a reactivity excursion. The analysis assumed that a fully inserted control rod becomes stuck in this position and the control rod drive was uncoupled and withdrawn. The rod subsequently becomes free and rapidly falls out of the core onto the withdrawn drive coupling. The amount of positive reactivity introduced into the reactor core was at a rate consistent with the maximum control rod drop velocity, resulting in the insertion of a large positive reactivity and a localized power excursion. The licensee's analysis of the AST CRDA had the occurrence of fuel damage and localized damage to fuel cladding, with a limited amount of melted fuel in the damaged rods.

The CRDA was based on a GE 10x10 fuel assembly in an 87.33 equivalent fuel pin array. It was assumed that 1,200 fuel rods were breached with 0.77 percent of melted fuel contained in the breached rods. A conservative radial peaking factor of 1.7 was used, and of the 0.77 percent of the melted fuel, 100 percent of noble gases and 50 percent of the iodine's contained in the melted fuel fraction were released to the reactor coolant. The source term used for the CRDA analysis was based on a power level of 3,528 megawatts thermal (MWt) and combined the release of the gap activity from the damaged fuel rods and the melted fuel. The gap activity and the activity from melted fuel mixed instantaneously in the reactor coolant within the reactor pressure vessel with no credit for partitioning or removal by the steam separators.

The following release scenarios were analyzed in the AST CRDA:

- The main condenser was assumed to leak activity into the turbine building at a rate of 1 percent per day. This activity is then released, unfiltered, to the environment by way of the reactor building/turbine building exhaust ventilation stacks, taking no credit for holdup in the Turbine Building.
- During power operation, the SJAE discharges to the offgas system.
- Flow is emitted from the sealing steam exhauster.
- The MVP is in operation during low power operation.
- 100 percent of the noble gases, 10 percent of the iodine species, and 1 percent of the particulate radionuclides were released from the turbine and condenser to the environment at a leak rate of 1 percent per day, for a period of 24 hours.

Licensee's Assessment of NEDO-31400A Conditions

As discussed above in SE Section 2.1, the NRC staff's SE for NEDO-31400 stated that licensees may reference the topical report as justification for proposing TS changes concerning the elimination of the MSLRM high radiation trip functions, provided that three conditions are met. The following provides the licensee's assessment of the three conditions associated with use of NEDO-31400A as described in Attachment 1 to the application dated September 3, 2014:

Condition 1

The applicant demonstrates that the assumptions with regard to input values (including power per assembly, Chi/Q, and decay times) that are made in the generic analysis bound those for the plant.

Although Condition 1 input values are not fully bounded, as-built information and the following assumptions were used, which demonstrate acceptable dose results:

1. The number of Failed Fuel Rods is assumed to be 1,200 rods for bounding case of 10x10 GNF2 fuel type. An average power peaking factor of 1.7 per pin was assumed. 10% of the core noble gases and iodine, and 12% of the core alkali metals, are released from the fuel gap.

2. Five percent (5.0%) of the breached fuel is conservatively assumed to melt during the CRDA. 100% of the noble gases and 50% of the iodines contained in the melted fuel fraction are assumed to be released to the reactor coolant.

3. The activity released from the breached fuel gap and melted fuel is assumed to be instantaneously mixed with the reactor coolant within the pressure vessel. 100% of all noble gases, 10% of the iodines, and 1% of remaining nuclides are transported to the Turbine/Main Condenser.

4. Upon detection of high radiation levels during a CRDA by the MSLRM, no credit is taken for MSIV closure, nor SJAE shutdown. Credit is taken for MVP cessation.

5. The Offgas System charcoal delay beds provide a retention time of 401 hours for xenon holdup and 34 hours for krypton holdup.

The analysis provided in NEDO-31400A assumes that the CRDA results in the failure of 850 fuel rods with a mass fraction of fuel in the damaged rods of 0.77%. For the portion of the fuel which was assumed to reach the melting point, the release fractions were 100% of the noble gases and 50% of the iodines.

The CRDA with no credit taken for MSIV closure is modeled with an EAB atmospheric dispersion factor (χ/Q) of 9.11E-4 sec/m^3. This model provides higher atmospheric dispersion than the analysis provided in NEDO-31400A for the scenario without MSIV closure, which models an EAB χ/Q of 3E-4 sec/m^3.

Condition 2

The applicant includes sufficient evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steam

lines will be controlled expeditiously to limit both occupational doses and environmental releases.

Response

Appropriate actions will be implemented at PBAPS to ensure that significant increases in MSL radiation levels are adequately controlled to limit occupational exposures and environmental releases. In the event of a MSLRM system high radiation alarm, MSLRM and Offgas System radiation level trending data from radiation monitor recorders will be reviewed, and if necessary, reactor coolant samples will be obtained and analyzed. If high radiation levels are confirmed, as measured by the Offgas system radiation monitors, reactor power will be reduced to maintain offgas release rates within TS requirements. If these release rates cannot be maintained within required TS limits, an orderly plant shutdown will be initiated. Plant procedures will be in place to implement the appropriate mitigative measures in response to a MSLRM high radiation alarm signal.

Condition 3

The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint at 1.5 times the normal Nitrogen-16 background dose rate at the monitor locations, and commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if the MSLRM or offgas radiation monitors or both exceed their alarm setpoints.

Response

The MSLRM alarm setpoint is currently set at 1.5 times the expected full reactor power background radiation level. Therefore, the MSLRM alarm setpoint does not need to be changed. As previously indicated in our response to item 2 above, samples will be taken, as necessary, to ascertain reactor coolant chemistry conditions and appropriate actions will be taken. Plant operators in the MCR will isolate the affected sample line PCIVs within one hour if plant conditions warrant.

NRC Staff Evaluation of Licensee's Conformance to NEDO-31400A

The following discusses the NRC's evaluation of the licensee's conformance to NEDO-31400A, Condition 1. This evaluation also includes evaluation of the additional changes proposed by the licensee that were beyond the scope of NEDO-31400A (i.e., proposed changes to eliminate the automatic closure of the MSL drain valves, MSL sample line valves, RHR system sample line valves, and reactor recirculation loop sample line valves).

In its application dated September 3, 2014, the licensee provided the following two calculations to support the proposed LAR:

- Calculation PM-1057, Revision 5, EAB, LPZ, and CR [control room] Doses due to Control Rod Drop Accident (CRDA) (Attachment 4 to the application)

- Calculation PM-1168, Revision 0, "Post-CRDA Release From RCS Sample Line" (Attachment 5 to the application)

The above calculations addressed the proposed new design-basis CRDA based on the changes proposed in the LAR. The NRC staff evaluated the assumptions, methods, and results of these calculations as discussed below.

The MSLRM system high radiation MCR alarms and trip function for isolating the MVP will be retained. This will ensure that any radioactive material released from a fuel failure will be contained in the main condenser and processed through the offgas treatment system. The offgas treatment system continuously removes non-condensable gases from the main condenser by the SJAEs during plant operation. The offgas treatment system reduces offgas radioactivity levels to permissible levels for release under all plant operations. The system uses catalytic recombination for volume reduction and control of hydrogen concentration and activated charcoal filters to adsorb fission product and activation gases prior to release to the environment. Instrumentation permits system operation and monitoring from the MCR.

The licensee has proposed the elimination of the MSLRM system high radiation automatic closure function for the MSL drain valves because the flow ultimately discharges into the main condenser, just as the flow from the MSIVs. Therefore, any radioactive material passing through the MSL drain valves to the main condenser and through the offgas treatment system is treated identically to any radioactive material that would pass through the MSIVs. The analysis in NEDO-31400A evaluated removing the MSLRM system high radiation trip function for closing the MSIVs. This same analysis applies for closure of the MSL drain valves. The elimination of the automatic closure of the MSL sample line valves, RHR system sample line valves, and reactor recirculation loop sample line valves is based on the fact that the effects are negligible, since these lines are small in comparison to the size of the lines associated with the MSIVs and MSL drains. The sample lines are routed to a sample sink where inlet valves on the sample lines are normally closed. These inlet valves must be opened periodically to allow for chemistry sampling. The sample sinks are located in the reactor building and are under equipment ventilation hoods. The equipment ventilation hood exhaust is filtered prior to release to the environment. In the unlikely event that chemistry sampling is in progress and the valve is open, a minimal amount of radioactive material would be released to the environment from the sample flow path. The licensee calculated that the TEDE dose received at the EAB through this flow path over the worst 2-hour period would be 0.286 rem. This would be in addition to that received from the CRDA, which was calculated to be 2.04 rem.

The new proposed design-basis CRDA assumes the rapid removal of the highest worth control rod, resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. For the dose consequence analysis, it was assumed that 1,200 of the fuel rods in the core were damaged, with melting occurring in 5.0 percent of the damaged rods. A core average radial peaking factor of 1.7 was used in the analysis. For releases from the breached fuel, 10 percent of the core inventory of noble gases and iodine species, and 12 percent of the core inventory of alkali metals, are assumed to be in the fuel gap.

For the release attributed to fuel melting, 100 percent of the noble gases and 50 percent of the iodine species is released to the reactor coolant.

Instantaneous mixing of the activity released from the fuel in the reactor coolant is assumed, with 100 percent of the noble gases, 10 percent of the iodine species, and 1 percent of alkali metal nuclides that are released into the reactor coolant, reaching the turbine and condenser. Of this activity that reaches the turbine and condenser, 100 percent of the noble gases, 10 percent of the iodine species, and 1 percent of alkali metal nuclides are available for release to the environment. Upon detection of high radiation levels during the CRDA by the MSLRM, no credit is taken for MSIV closure nor SJAE shutdown, but credit is taken for MVP shutdown if it is in operation. The main condenser is assumed to leak activity into the atmosphere from the turbine building/reactor building ventilation exhaust stack at a rate of 1 percent of the condenser volume per day, for a period of 24 hours. This leaked activity is then released unfiltered into the environment by way of the turbine building/reactor building ventilation exhaust stack, taking no credit for holdup in the turbine building. The forced flow path through the SJAEs discharges to the off-gas system. This pathway credits elimination of iodine releases and delay of noble gas releases by the off-gas system charcoal delay beds. The licensee assumed that chemistry sampling is in progress, of either the RHR system sample line or reactor recirculation loop sample line, and that purging of the sample line continues for 60 minutes until the sample valves are closed by a control room operator.

The PBAPS, Unit 2 and 3, control room is modeled as a closed volume of 176,000 cubic feet. The control room was modeled without taking credit for automatic system actuation. Although the normal maximum flow into the control room is 20,600 cubic feet per minute (cfm), it was assumed that there is an additional unfiltered inleakage rate of 500 cfm. No credit is taken for any filtration of intake flow into the control room.

Exelon evaluated the radiological consequences resulting from the postulated new proposed design-basis CRDA and concluded that the radiological consequences at the EAB, LPZ, and MCR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP 15.0.1. The licensee's calculated dose results are given in Table 1.

During review of the information submitted by the licensee, the NRC staff noticed that the licensee's application dated September 3, 2014, states as follows on page 13 of Attachment 1:

> In order to assess the radiological impact of a scenario where a CRDA occurred coincidentally with an open sample line, it was conservatively assumed that the largest sample line is "open" for one hour prior to being isolated by remote-manual action taken by licensed MCR operators.

Exelon's calculation number PM-1168 assumes a sample purge rate of 500 milliliters per minute (mL/min) or 500 cubic centimeters per minute, but does not provide the maximum flow rate possible for each of the sample lines. The NRC staff requested that the licensee provide the maximum flow rate possible from the MSL sample line, RHR sample line, and the reactor recirculation loop sample line and discuss any included safety margin included in the chosen sample purge rate of 500 milliliters per minute, or 500 cubic centimeters per minute. In addition, the licensee was requested to explain if it is possible to have all three sample lines open at the same time, and if so, explain how this is accounted for in Calculation PM-1168.

The licensee responded in the supplement dated February 9, 2015, as follows:

> A nominal flushing flow rate of 500 mL/min was chosen for the sample flowrate based on instructions in the applicable chemistry sampling procedure. The sample stations are flushed at various flowrates; however, 500 mL/min is the largest. This value, when combined with the assumed time for operators to isolate the sample line, provides the total volume of water which is released. This total value is considered to be conservative; however, to ensure the calculation is bounding of the Chemistry sampling procedure, a maximum limit of 700 mL/min will be established in the Chemistry procedure by this modification. The calculation has been revised to reflect the new assumed flow value of 700 mL/min and that the assumed isolation time is 40 minutes. In order to maintain the same total volume release, the assumed isolation time will be reduced to 40 minutes, which still provides sufficient time for isolation. This results in a release of 28,000 cc; however, 30,000 cc will be utilized for conservatism. This does not affect the results of this calculation.
>
> Additionally, conservatism is established in this portion of the calculation by assuming that the water coming from the sample line instantly reaches the temperature of the RCS due to failure of the non-safety related sample line chiller.
>
> Procedural controls are in place to prevent the applicable sample lines from being open at the same time. Normal Chemistry operation involves having the Reactor Water Clean-Up (RWCU) influent sample line open at the RWCU sample station. The MSL sample line, RHR sample line and Reactor Recirculation sample line would be closed. Chemistry procedure CH-407, "Sampling of Reactor Water," establishes the controls and provides the guidance for obtaining samples from RWCU influent (at RWCU sample station), Reactor Recirculation and RHR heat exchanger outlet, which are both at the Reactor Building sample station (commonly referred to as the Feed Water sample station).

Furthermore, in evaluating the submittal, the NRC staff could not determine if RG 1.183, Section 4.4, "Acceptance Criteria," had been assessed and met for PBAPS. The NRC staff requested that the licensee provide additional information describing what was meant by remote-manual action to isolate a sample line and discuss whether or not the operator action of isolating the sample line would meet the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," Task Action II.B.2, November 1980 (ADAMS Accession No. ML051400209). These NUREG-0737 requirements ensure that these actions can be completed without exceeding the acceptance criteria (typically contained in final GDC 19) for mission doses. The licensee response stated that the sample lines are isolated from the MCR, rather than locally at the sample valve itself. Since operators would not be required to leave the MCR, NUREG-0737, Task Action II.B.2, is not applicable for the operator action of isolating the sample line.

Exelon's Calculation PM-1057, pathway 1 (reactor coolant to the condenser) shows that flow from the reactor coolant system stops at 10 minutes. However, without the MSLRM and its automatic MSIV closure, the MSIVs do not close, and steam flow should continue for 24 hours before isolation. The NRC staff requested that the licensee explain this discrepancy, and if needed, provide the new analysis showing that the MSIVs remain open. In the supplement dated February 9, 2015, the licensee responded that, as described in Sections 2.3.1 and 7.3 of Calculation PM-1057, the release is modeled such that 99 percent of the coolant activity will be transferred to the main condenser within 1 second, which conservatively models the release to the main condenser, and that this transfer method has been used since the AST was approved for use at PBAPS. The NRC staff finds the licensee's response acceptable.

Based on review of the licensee's application dated September 3, 2014, and the supplement dated February 9, 2015, the NRC staff finds:

1) The licensee's assumptions and inputs used in its analysis for the proposed new design-basis CRDA are consistent with the guidance in RG. 1.183. The assumptions found acceptable to the NRC staff are presented in Table 2 of this SE. The licensee's calculated dose results are given in Table 1 of this SE. The dose results meet the NRC's acceptance criteria for a CRDA of 6.3 rem at the EAB for the maximum 2-hour period, 6.3 rem at the outer boundary of the LPZ during the entire period of the postulated cloud passage, and 5 rem in the MCR for the duration of the accident.

2) The licensee has provided sufficient evidence to provide reasonable assurance that appropriate actions will be implemented at PBAPS to ensure that significant increases in MSL radiation levels are adequately controlled to limit occupational exposures and environmental releases.

3) The licensee has established an appropriate MSLRM alarm setpoint (i.e., 1.5 times the full reactor power normal background radiation level). In addition, in the event of a high MSLRM alarm, MSLRM and offgas system radiation level trending data will be reviewed and, if necessary, reactor coolant samples will be obtained and analyzed. If high radiation levels are confirmed, appropriate actions will be taken in accordance with the plant TS requirements.

Based on the above findings, the NRC staff concludes that the proposed LAR meets the intent of the three conditions in the NRC's SE for NEDO-31400A, to justify eliminating the MSLRM trip and isolation functions from initiating a reactor scram and automatic closure of the MSIVs. In addition, the licensee's analysis also justifies the proposed changes to eliminate the automatic closure of the MSL drain valves, MSL sample line valves, RHR system sample line valves, and reactor recirculation loop sample line valves.

The NRC staff performed independent confirmatory dose evaluations, as needed, to ensure a complete understanding of the licensee's methods. The staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and MCR doses for the new proposed design-basis CRDA will continue to comply with the applicable dose criteria as shown in Table 1 of this SE.

Based on the above evaluation, the NRC staff finds the proposed LAR acceptable with respect to radiological consequences.

Table 1
PBAPS Units 2 and 3 Radiological Consequences Expressed as TEDE (rem)

DBA	EAB	LPZ	MCR
CRDA(1) without reactor water sample release	2.04	0.35	1.77
Reactor Water Sample Release	0.286	0.043	0.290
CRDA(1) with reactor water sample release	2.33	0.39	2.06
Dose Criteria	6.3	6.3	5.0

(1) 1200 failed rods at full power

Table 2
PBAPS Units 2 and 3 AST Data and Assumptions for the CRDA

Core Thermal Power Level	4030 MWt
Radial Peaking Factor	1.7
Number of fuel assemblies in core	764
Number of equivalent fuel rods per assembly - GNF2 10 x 10	85.6
Number of fuel rods damaged in full power CRDA	1200
Fraction of fission product inventory in gap	
Noble gases	10%
Iodines	10%
Alkali Metals (Cs and Rb)	12%
Fraction of damaged rods experiencing fuel melt	5.0%
Fraction of activity in melted regions released to RCS	
Noble gases	100%
Iodines	50%
Others as specified by	Table 1 of RG 1.183
Fraction of activity release in RCS reaching condenser	
Iodines	10%
Noble gases	100%
Others	1%
Fraction of activity from condenser for release to environment	
Iodines	10%
Noble gases	100%
Others	1%
Release rate from condenser to turbine building	1% per day for 24 hours
Release rate from gland seal condenser to turbine building	18,920 lbs/hr (0.15% of main steam flow)
MCR isolation and MCR emergency ventilation initiation	Not credited
MCR normal flow intake rate, plus an unfiltered inleakage flow rate, is used with and control room occupancy factor for 24 hour duration of accident.	1

3.4 Technical Evaluation Conclusion

Based on the evaluation in SE Sections 3.1, 3.2, and 3.3, the NRC staff concludes that the proposed amendments are 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 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 (79 FR 67201). 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.

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Date: July 28, 2015

July 28, 2015

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: ELIMINATE MAIN STEAM LINE RADIATION MONITOR TRIP AND ISOLATION FUNCTION (TAC NOS. MF4757 AND MF4758)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment Nos. 299 and 302 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the 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 September 3, 2014, as supplemented by letter dated February 9, 2015.

The amendments revise the TSs to eliminate the main steam line (MSL) radiation monitor from initiating: (1) a reactor protection system automatic reactor scram; and (2) a primary containment isolation system isolation, including automatic closure of the MSL isolation valves, MSL drain valves, MSL sample line valves, residual heat removal system sample line valves, and reactor recirculation loop sample line valves.

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. 299 to Renewed DPR-44
2. Amendment No. 302 to Renewed DPR-56
3. Safety Evaluation

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