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

August 28, 2019

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
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 AMENDMENT NOS. 327 AND 330 RE: REDUCE HIGH PRESSURE SERVICE WATER SYSTEM DESIGN PRESSURE AND TEMPORARILY EXTEND COMPLETION TIMES (EPID L-2018-LLA-0265)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 327 and 330 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3, respectively. These amendments consist of changes to Technical Specifications 3.6.2.3, 3.6.2.4, 3.6.2.5, and 3.7.1 in response to your application dated September 28, 2018, as supplemented by letters dated February 15, 2019; March 26, 2019; and May 23, 2019.

The amendments authorize revisions to the design and licensing basis described in the Updated Final Safety Analysis Report to reduce the design pressure rating of the high pressure service water (HPSW) system. This change provides additional corrosion margin in the HPSW system pipe wall thickness, increasing the margin of safety for the existing piping. In addition, this change also temporarily revises certain Technical Specifications to allow sufficient time to perform modifications of the HPSW system to support the proposed reduction of the HPSW design pressure and to allow for timely repairs of a heat exchanger at Peach Bottom, Unit 3.

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

Sincerely,

Jenny Tobin

Jennifer C. Tobin, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 327 to Renewed DPR-44
2. Amendment No. 330 to Renewed DPR-56
3. Safety Evaluation

cc: 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. 327
Renewed License No. DPR-44

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

   A. The application for amendment by Exelon Generation Company, LLC and PSEG Nuclear LLC (the licensees), dated September 28, 2018, as supplemented by letters dated February 15, 2019; March 26, 2019; and May 23, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the 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. 327, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 28, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 327

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

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
3.6-27	3.6-27
3.6-29	3.6-29
3.6-30a	3.6-30a
3.7-1	3.7-1

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

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

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the Peach Bottom Atomic Power Station, Unit 2, at steady state reactor core power levels not in excess of 4016 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 327, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

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

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

(4) Fire Protection

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

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

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days*
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. Two RHR suppression pool cooling subsystems inoperable.	C.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

* The 7-day Completion Time for one RHR suppression pool cooling subsystem inoperable may be extended to 10 days four (4) times until December 31, 2021 with compensatory measures identified in EGC License Amendment Request letter dated September 28, 2018 established and in effect, to allow for modifications to the HPSW system.

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool spray subsystem inoperable.	A.1 Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days*
B. Two RHR suppression pool spray subsystems inoperable.	B.1 Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

* The 7-day Completion Time for one RHR suppression pool spray subsystem inoperable may be extended to 10 days four (4) times until December 31, 2021 with compensatory measures identified in EGC License Amendment Request letter dated September 28, 2018 established and in effect, to allow for modifications to the HPSW system.

3.6 CONTAINMENT SYSTEMS

3.6.2.5 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.2.5 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1	Restore RHR drywell spray subsystem to OPERABLE status.	7 days*
B. Two RHR drywell spray subsystems inoperable.	B.1	Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

* The 7-day Completion Time for one RHR drywell spray subsystem inoperable may be extended to 10 days four (4) times until December 31, 2021 with compensatory measures identified in EGC License Amendment Request letter dated September 28, 2018 established and in effect, to allow for modifications to the HPSW system.

3.7 PLANT SYSTEMS

3.7.1 High Pressure Service Water (HPSW) System

LCO 3.7.1 Two HPSW subsystems and the HPSW cross tie shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One HPSW subsystem inoperable.	-----------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for RHR shutdown cooling made inoperable by HPSW System. ----------------------------	
	A.1 Restore HPSW subsystem to OPERABLE status.	7 days*
B. HPSW cross tie inoperable.	B.1 Restore HPSW cross tie to OPERABLE status.	7 days*
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours

(continued)

* The 7-day Completion Time for one HPSW subsystem inoperable may be extended to 10 days four (4) times until December 31, 2021 and the 7-day Completion Time for the HPSW cross tie inoperable may be extended to 10 days two (2) times until December 31, 2021 with compensatory measures identified in EGC License Amendment Request letter dated September 28, 2018 established and in effect, to allow for modifications to the HPSW system.

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. 330
Renewed License No. DPR-56

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

   A. The application for amendment by Exelon Generation Company, LLC and PSEG Nuclear LLC (the licensees), dated September 28, 2018, as supplemented by letters dated February 15, 2019; March 26, 2019; and May 23, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 330, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 28, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 330

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

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
3.6-27	3.6-27
3.6-29	3.6-29
3.6-30a	3.6-30a
3.7-1	3.7-1

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

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

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the Peach Bottom Atomic Power Station, Unit No. 3, at steady state reactor core power levels not in excess of 4016 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 330, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

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

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

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

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days*
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. Two RHR suppression pool cooling subsystems inoperable.	C.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

* The 7-day Completion Time for one RHR suppression pool cooling subsystem inoperable may be extended to 10 days (3) times and to 14 days one (1) time (A-C subsystem only) until December 31, 2021 with compensatory measures identified in EGC License Amendment Request letter dated September 28, 2018 established and in effect, to allow for modifications to the HPSW system and repairs to Unit 3 RHR Heat Exchanger 3CE024.

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool spray subsystem inoperable.	A.1 Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days*
B. Two RHR suppression pool spray subsystems inoperable.	B.1 Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

* The 7-day Completion Time for one RHR suppression pool spray subsystem inoperable may be extended to 10 days three (3) times and 14 days one (1) time (A-C subsystem only) until December 31, 2021 with compensatory measures identified in EGC License Amendment Request letter dated September 28, 2018 established and in effect, to allow for modifications to the HPSW system and repairs to Unit 3 RHR Heat Exchanger 3CE024.

3.6 CONTAINMENT SYSTEMS

3.6.2.5 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.2.5 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1	Restore RHR drywell spray subsystem to OPERABLE status.	7 days*
B. Two RHR drywell spray subsystems inoperable.	B.1	Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

* The 7-day Completion Time for one RHR drywell spray subsystem inoperable may be extended to 10 days three (3) times and 14 days one (1) time (A-C subsystem only) until December 31, 2021 with compensatory measures identified in EGC License Amendment Request letter dated September 28, 2018 established and in effect, to allow for modifications to the HPSW system and repairs to Unit 3 RHR Heat Exchanger 3CE024.

3.7 PLANT SYSTEMS

3.7.1 High Pressure Service Water (HPSW) System

LCO 3.7.1 Two HPSW subsystems and the HPSW cross tie shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One HPSW subsystem inoperable.	-----------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for RHR shutdown cooling made inoperable by HPSW System. ---------------------------	
	A.1 Restore HPSW subsystem to OPERABLE status.	7 days*
B. HPSW cross tie inoperable.	B.1 Restore HPSW cross tie to OPERABLE status	7 days*
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours

(continued)

* The 7-day Completion Time for one HPSW subsystem inoperable may be extended to 10 days three (3) times and 14 days one (1) time until December 31, 2021; and the 7-day Completion Time for HPSW cross tie inoperable may be extended to 10 days one (1) time and 14 days one (1) time (A-C subsystems only) until December 31, 2021 with compensatory measures identified in EGC License Amendment Request letter dated September 28, 2018 established and in effect, to allow for modifications to the HPSW system and repairs to Unit 3 RHR Heat Exchanger 3CE024.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 327 AND 330

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 28, 2018, as supplemented by letters dated February 15, 2019; March 26, 2019; and May 23, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18275A023, ML19046A129, ML19085A385, and ML19143A176, respectively) Exelon Generation Company LLC (Exelon or the licensee) submitted a request for changes to the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3, Technical Specifications (TSs).

These amendments would authorize revisions to the design and licensing basis described in the Updated Final Safety Analysis Report (UFSAR) to reduce the design pressure rating of the high pressure service water (HPSW) system. This change would provide additional corrosion margin in the HPSW system pipe wall thickness, thereby increasing the margin of safety for the existing piping. In addition, this change would also temporarily revise certain TSs to allow sufficient time to perform modifications of the HPSW system to support the proposed reduction of the HPSW design pressure and to allow for timely repairs of a heat exchanger at Peach Bottom, Unit 3.

2.0 REGULATORY EVALUATION

2.1 HPSW System Functions

As described in the Peach Bottom UFSAR and the license amendment request (LAR), the HPSW system removes the heat rejected by the residual heat removal system (RHR) heat exchangers during shutdown operation and accident conditions. The major flow paths of the HPSW system consist of two independent parallel flow loops serving each unit, designated Division I and Division II, respectively. Each flow loop (i.e., subsystem) contains two HPSW pumps that discharge to a common header serving two RHR heat exchangers connected in parallel. Division I contains HPSW pumps A and C, and Division II contains HPSW pumps B and D. In the normal open loop alignment, the HPSW pumps take suction from the Conowingo Pond (ultimate heat sink) through the service water bay, and the HPSW loops discharge through a common pipe for each unit to the discharge pond. The discharge pipe contains a normally open motor-operated isolation valve (MOV) to the pond and a pipe connection with a normally closed MOV valve to the emergency cooling water system to provide an alternate discharge in the event that the Conowingo Dam fails or the pond floods.

Each unit has four HPSW pumps currently configured to deliver 4,500 gallons per minute (gpm) each at a total developed head of about 303 pressure differential (psid) and with a shutoff head of about 444 psid at zero flow. The HPSW six-stage vertical pumps are driven by 1,000 horsepower electric motors. Each unit has four RHR vertically-oriented shell and tube heat exchangers with HPSW flow through the tube side and RHR flow through the shell side. The heat exchanger design pressure is 450 pressure gauge (psig) on both the tube and shell sides.

The RHR heat exchanger HPSW outlet motor-operated throttle valves (MOV) are normally closed when the HPSW system is in standby with idle pumps. When establishing HPSW flow to an RHR heat exchanger, the associated outlet throttle valve is opened and then the HPSW pump is started. The valves are currently throttled as necessary to achieve the required HPSW flow rate and to maintain HPSW pressure at least 20 psid higher than RHR pressure. At maximum expected RHR system pressures, a pressure of 233 psig at the HPSW pump discharge is required to ensure HPSW side pressure exceeds RHR system pressure at the RHR heat exchanger. HPSW flow restricting orifices are installed downstream of the throttling valves to suppress cavitation in the throttle valves by reducing the required pressure drop across the throttled valves.

A divisional cross-tie line connecting the two HPSW loops on each unit is provided with a normally closed MOV. The divisional cross-tie is provided with redundant power supplies and is required to be operable during Modes 1, 2, and 3 to allow HPSW pumps from one subsystem to supply RHR heat exchangers in the opposite loop, if required. A unit cross-tie line with two normally closed manual isolation valves is also provided between one Unit 2 HPSW subsystem (B/D) and one Unit 3 HPSW subsystem (B/D). The cross-tie lines provide the flexibility to establish alternate flow alignments, if needed, under emergency conditions. A supply connection from the HPSW system to the RHR system, through two normally closed MOVs, is provided from one HPSW subsystem per unit to permit the HPSW system furnishing a backup water supply to RHR for post-accident containment flooding.

The RHR and HPSW systems are currently designed such that HPSW operates at a higher pressure than RHR at the RHR heat exchanger interface; however, during standby conditions, the RHR system pressure is maintained greater than HPSW. The RHR and HPSW systems are standby systems that typically operate during testing or plant shutdown. With this design, if there is an internal leak within an RHR heat exchanger during standby conditions, RHR water, which is normally torus (containment pressure suppression pool) water, leaks into the HPSW system until the HPSW system pressure exceeds that of the RHR system. The HPSW system, when in operation, is maintained at a higher pressure than the RHR system at the RHR heat exchanger interface, which prevents RHR water leakage into HPSW.

Each RHR heat exchanger currently contains a tube-to-shell differential pressure alarm, which alerts the operator if there is insufficient differential pressure between HPSW and RHR when the associated RHR heat exchanger HPSW outlet throttle valve is not closed. Additionally, there are radiation monitors that sample the HPSW system both upstream and downstream of the RHR heat exchangers to indicate if there is cross-system leakage. These alarms and established operations and chemistry procedures are used to identify and respond to any potential RHR/HPSW system cross-system leakage issues.

2.2 Proposed Design-Basis Changes

The proposed amendments would revise the Peach Bottom, Units 2 and 3, design and licensing basis described in the UFSAR to reduce the design pressure rating of the HPSW system. The current design pressure rating of the HPSW system supply piping from the HPSW pumps to, and including, the RHR heat exchanger HPSW outlet MOV will be reduced from 450 psig to a new design pressure rating of approximately 200 psig. The final value is expected to fall between 190 psig and 230 psig and be verified by system operation with required flow rates after the physical changes are completed to establish the new normal system operating pressure.

The existing RHR heat exchangers located in the affected piping have a shell and tube design pressure rating of 450 psig. The pressure rating of these heat exchangers will remain unchanged. After the proposed changes in system design, the operating pressure of the HPSW system will be reduced to a value below the maximum RHR system operating pressure.

The design pressure reduction will be achieved through physical plant modifications to reduce HPSW operating pressure. These include modifications to HPSW pumps to produce a lower discharge pressure at the desired flow rate, replacement of RHR heat exchanger HPSW outlet flow control throttling valves, and removal of flow restricting orifices downstream of the throttling valves. The design pressure reduction will be achieved through physical modifications to reduce the operating pressure of the system, which include the following:

- Replace or modify the HPSW pumps to deliver lower head (all four pumps on both units). Piping design pressures are reduced with the reduction in pump shutoff head and maximum sustained operating pressure.

- Replace the HPSW outlet valves to achieve higher flow capacity (less pressure drop) with cavitation-resistant trim.

- Remove flow restricting orifices in the HPSW piping downstream of the RHR heat exchanger HPSW outlet valves. These orifices currently generate a significant portion of the system hydraulic resistance. For HPSW subsystems with multiple restricting orifices (Units 2 and 3 loops A and D), the design pressure of the modified piping in this part of the system will also be reduced from 300 psig to 150 psig, consistent with the design pressure of the existing downstream piping.

There will be no change to the current licensing basis, as reflected in the UFSAR requirement for HPSW flow delivery to the RHR heat exchangers for post-accident containment cooling or other design-basis functions. The physical modifications required to reduce HPSW system design pressure are expected to be installed with the plant online in Mode 1.

2.3 Proposed Technical Specification Changes

The proposed changes will temporarily revise the Peach Bottom, Units 2 and 3, TS Sections 6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling"; TS 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray"; TS 3.6.2.5, "Residual Heat Removal (RHR) Drywell Spray"; and TS 3.7.1, "High Pressure Service Water (HPSW) System."

TS 3.6.2.3 will allow, on a one-time basis for each subsystem, the Required Action A.1 for one RHR suppression pool cooling subsystem inoperable completion time (CT) to be extended from 7 days to 10 days for Unit 2 and the same for Unit 3 subsystems, except that the RHR suppression pool cooling A-C subsystem Action A.1 CT would be extended from 7 days to 14 days to allow for repair of the Unit 3 RHR Heat Exchanger 3CE024.

TS 3.6.2.4 will allow, on a one-time basis for each subsystem, the Required Action A.1 for one RHR suppression pool spray subsystem inoperable CT to be extended from 7 days to 10 days for Unit 2 and the same for Unit 3 subsystems, except that the RHR suppression pool spray A-C subsystem Action A.1 CT would be extended from 7 days to 14 days to allow for repair of the Unit 3 RHR Heat Exchanger 3CE024.

TS 3.6.2.5 will allow, on a one-time basis for each subsystem, the Required Action A.1 for one RHR drywell spray subsystem inoperable CT to be extended from 7 days to 10 days for Unit 2 and the same for Unit 3 subsystems, except that the RHR drywell spray A-C subsystem action A.1 CT would be extended from 7 days to 14 days to allow for repair of the Unit 3 RHR Heat Exchanger 3CE024.

TS 3.7.1 will allow, on a one-time basis for each subsystem, the Required Action A.1 for one HPSW subsystem inoperable CT to be extended from 7 days to 10 days for Unit 2 and the same for Unit 3 subsystems, except that the HPSW A-C subsystem action A.1 CT would be extended from 7 days to 14 days to allow for repair of the Unit 3 RHR Heat Exchanger 3CE024. Also, TS 3.7.1 will allow two CT extensions of Required Action B.1 for the HPSW cross-tie inoperable from 7 days to 10 days for Unit 2 and one allowable CT of Required Action B.1 for the HPSW cross-tie inoperable from 7 days to 10 days for Unit 3, with an additional one CT extension from 7 days to 14 days for the A-C subsystem only to allow for repairs to the Unit 3 RHR Heat Exchanger 3CE024.

The requested changes to TS 3.6.2.3, TS 3.6.2.4, TS 3.6.2.5, and TS 3.7.1 also include a note stating that the CT extensions are allowed until December 31, 2021, with compensatory measures identified in the licensee's application dated September 28, 2018, established and in effect to allow for modifications to the HPSW system and for Unit 3 to also allow for repairs to the Unit 3 RHR Heat Exchanger 3CE024.

The Peach Bottom, Units 2 and 3, HPSW systems have exhibited a history of degradation similar to raw fresh water systems throughout the nuclear industry. In March 2014, Exelon submitted a request to the NRC in accordance with 10 CFR 50.55a(a)(3)(i). This request proposed an alternative from the American Society of Mechanical Engineers (ASME) Code, Section XI, IWD-3120(b) requirement to perform repair/replacement activities for degraded HPSW piping, which has a maximum operating pressure in excess of 275 psig. The proposed alternative was to apply the evaluation methods of ASME Code Case N-513-3, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1," to the Class 3 HPSW system piping having a maximum operating pressure greater than 275 psig but less than or equal to 375 psig. The objective was to avoid accruing additional personnel radiation exposure and increased plant risk associated with a plant shutdown to comply with ASME Code, Section XI requirements. In addition, the use of an acceptable alternative analysis method in lieu of immediate action for a degraded condition allowed Exelon to perform additional extent of condition examinations on the affected systems, while allowing time for safe and orderly long-term repair actions if necessary. The licensee's installation plan requires that the HPSW modifications be performed in phases, using multiple entries into the TS limiting conditions for operation (LCOs) action statements.

The NRC approved the licensee's relief request (RR 14R-55) on September 6, 2016 (ADAMS Accession No. ML16230A237). Exelon is currently implementing the monitoring requirements of ASME Code Case N-513-3 at Peach Bottom, Units 2 and 3, with an allowable leak rate of 5 gpm for the HPSW system piping. The NRC authorized use of RR 14R-55 until the end of the fourth inservice inspection (ISI) interval, until the acceptance criteria of Code Case N-513-3 are exceeded, or until the leak rate exceeds the allowable limit, whichever occurs first. The fourth ISI interval ended on December 31, 2018. Exelon submitted an additional relief request on March 26, 2018 (ADAMS Accession No. ML18086B110), to address the fifth ISI interval. On December 21, 2018 (ADAMS Accession No. ML18327A062), the NRC authorized continued use of Code Case N-513-3 for the fifth ISI interval, which ends on December 31, 2028.

2.4 Description of the Regulatory Requirements

The licensee for Peach Bottom has made changes to the facility over the life of the plant that 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 construction permit for Peach Bottom was issued by the Atomic Energy Commission (AEC) on January 31, 1968. As discussed in Appendix H to the Peach Bottom UFSAR, during the construction/licensing process, Peach Bottom was 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 (the draft GDC). Appendix H to the Peach Bottom UFSAR contains an evaluation of the design basis of Peach Bottom against the draft GDC. The licensee concluded that Peach Bottom conforms 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" (the final GDC). Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the U.S. Nuclear Regulatory Commission (NRC or the Commission) Staff Requirements Memorandum, SECY-92-223, "Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML12256B290), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. At the time of the promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the final GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the final GDC were formally adopted was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission.

Criterion 16 specifies that "Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage." The HPSW system is designed with radiation monitors to detect potential radioactive releases to the environment from an RHR heat exchanger leak.

Criterion 17 specifies that "Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions."

Criterion 37 specifies that:

> Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Criterion 38 specifies that:

> All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Criterion 52 specifies that "Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided."

Criterion 70 specifies that:

> The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR Part 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR Part 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

The regulation at 10 CFR 50.36, "Technical specifications," states that the TSs include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. Section 50.36 of 10 CFR also states that when an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition is met. Section 50.36 of 10 CFR does not specify what actions are required or how quickly they must be completed.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals complies with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(b), 12 months prior to the start of the 120-month interval, subject to the conditions listed therein. The ASME Code, Section XI, IWA-4000, "Repair/Replacement Activities," provides provisions for the replacement activities, which are applied to the proposed LAR.

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," the licensee proposed changes to the Appendix A TSs of Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom, Units 2 and 3, respectively. The NRC staff evaluated the licensee's new design and proposed TS changes to determine if they affect the radiological consequences for the design-basis accidents, and confirm that the radiological dose requirements specified in 10 CFR 50.67(b)(2) and dose limits specified in 10 CFR Part 50, Appendix A, GDC 19, are met.

For licensees that use the alternative source term in their dose consequence analyses, the NRC staff uses the regulatory guidance provided in NUREG-0800, Revision 0, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," dated July 2000 (ADAMS Accession No. ML003734190), and the methodology and assumptions stated in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792). The NRC staff also considered relevant information in the Peach Bottom UFSAR.

3.0 TECHNICAL EVALUATION

HPSW Function

The safety function of the RHR system is to restore and maintain the coolant inventory in the reactor vessel so that the core is adequately cooled after a loss-of-coolant accident (LOCA). It also provides cooling for the containment so that condensation of the steam resulting from the blowdown due to the design-basis LOCA is ensured.

RHR has multiple modes of operation including shutdown cooling, containment cooling, and low pressure coolant injection. In the shutdown cooling mode, it is capable of completing a normal cooldown to 125 degrees Fahrenheit (°F) (52 degrees Celsius °C)) in about 35 hours (with two shutdown cooling subsystems in service) after the main condenser is no longer available. In this mode, the RHR system pumps reactor coolant from one of the recirculation loops through the RHR heat exchangers where cooling takes place by transferring heat to the HPSW system. It is returned to the reactor vessel by either recirculation loop.

In the containment cooling mode, the RHR system provides a means to cool the containment either through suppression pool cooling or containment spray. The safety-related function of suppression pool cooling is to remove reactor core decay heat and sensible heat discharged to the suppression pool after a design-basis event or accident in order to maintain the suppression pool temperature within an acceptable limit and containment pressure within an acceptable range. When in the containment cooling mode, the RHR pumps are aligned to pump water from the suppression pool through the heat exchangers. It is then either returned to the suppression pool by the full flow test line or diverted to the spray headers in the drywell and above the suppression pool.

In the low pressure coolant injection mode, the RHR system operates with the high pressure coolant injection core spray and automatic depressurization systems to restore and, if necessary, maintain the coolant inventory in the reactor vessel after a LOCA so that the core is sufficiently cooled to preclude excessive fuel clad temperatures and subsequent energy release due to a metal-water reaction. During low pressure coolant injection operation, the RHR pumps take suction from the suppression pool and discharge into the core region of the reactor vessel through the recirculation loops.

HPSW System Pressure

In the LAR, the NRC staff noted that various pressures (190, 200, 230, 233, and 240 psig) were provided as the final design pressure. In a request for additional information dated January 16, 2019 (ADAMS Accession No. ML19017A047), the NRC staff requested Exelon to provide the exact design pressure and maximum operating pressure in the HPSW piping system after the proposed modification and discuss the pressure that is used to calculate the corrosion margin.

In response to the NRC's request dated January 16, 2019 (ADAMS Accession No. ML19046A129), the staff found that the licensee had clarified the corrosion rate calculation and provided the calculated corrosion rates at various HPSW pipe locations. The NRC staff determined that the calculated corrosion rates are consistent with industry operating experience. Exelon also provided appropriate information regarding the pipe size, pipe wall thickness, and final design pressure at various pipe locations. The NRC staff verified Exelon's calculation of minimum pipe wall thickness based on its independent calculations. In the response, the licensee also stated that the HPSW system has piping regions with different design pressures, which are reflective of the local operating conditions. However, there will be one HPSW system maximum design pressure, which is expected to be within the range of 200-230 psig. Exelon stated that at the time the LAR was submitted, a range of pressures was discussed in the request because the specific final design characteristics for the new low head pumps and the new valves downstream of the RHR heat exchangers had not yet been established. One specific design characteristic that will contribute to the exact maximum design pressure is the pump shutoff head. The final design documents and supporting calculations will establish the exact design pressure. This selected design pressure will be used to calculate the corrosion margin. In the March 26, 2019, supplement (ADAMS Accession No. ML19085A385), the licensee stated that Table 1 (of the supplement) shows the corrosion data at 230 psig. The minimum required wall thicknesses for a design pressure of 200 psig would be slightly lower than, or possibly the same as, those for 230 psig, depending upon the type of stress (pressure-related or non-pressure-related), and thus sets a conservative minimum required wall thickness.

NRC's Branch Technical Position BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" (ADAMS Accession No. ML070800027), specifies that high energy piping has a maximum operating pressure exceeding 275 psig, or maximum operating temperature exceeding 200 °F (93 °C). The NRC staff notes that if the final design pressure is reduced from 450 psig to 230 psig, the HPSW piping would be classified as moderate energy piping. With a lower design pressure, the allowable minimum pipe wall thickness would be lower; therefore, the margin between the measured pipe wall thickness and the allowable wall thickness would increase. The NRC staff finds that the higher margin in pipe wall thickness would result in increased flexibility and safety in plant operation.

The licensee indicated that the new design pressure rating of the HPSW supply piping will be reduced to approximately 200 psig. The final value is expected to fall between 190 psig and 230 psig in the detailed design with final vendor inputs. Exelon stated that calculations will be revised to establish the new piping minimum wall thickness requirements at the reduced design pressure. The design pressure reduction will be achieved through physical modifications to reduce the operating pressure of the system, which include the following:

- Replace or modify the HPSW pumps to deliver lower head (all four pumps on both units). Piping design pressures are reduced with the reduction in pump shutoff head and maximum sustained operating pressure.

- Replace MO-089 throttle valves to achieve higher flow capacity (less pressure drop) with cavitation-resistant trim.

- Remove flow restricting orifices in the HPSW piping downstream of the RHR heat exchanger MO-089 throttle valves. These orifices currently generate a significant portion of the system hydraulic resistance. For HPSW subsystems with multiple restricting orifices (Units 2/3A and 2/3D), the design pressure of the modified piping in this part of the system will also be reduced from 300 psig to 150 psig, consistent with the design pressure of the existing downstream piping.

Additional modifications to HPSW system components such as the overpressure protection relief valves and pressure monitoring instrumentation will be made, as required, to accommodate the proposed change.

<u>HPSW System Flow and Pressure</u>

Exelon reported that there will be no change to the design-basis requirement for HPSW flow delivery to the RHR heat exchangers for post-accident containment cooling or other design-basis functions. The HPSW system on each unit contains two independent and redundant loops (subsystems) where each loop contains two HPSW pumps. Either loop is capable of providing the required post-accident containment cooling capacity with two HPSW pumps and two RHR heat exchangers.

The licensee also explained that the replacement HPSW pumps will be sized to deliver the required flow (with margin) at a reduced discharge pressure. Because there is no change to the design-basis HPSW flow delivery to the RHR heat exchangers and no change to HPSW supply temperature, the proposed change will have no impact on RHR heat exchanger thermal performance or containment heat removal for normal operation and design-basis events.

In the LAR, the NRC staff noted that with the proposed pressure reduction, the pipe stresses will be different from the original design basis for the HPSW piping. The NRC staff also noted that the LAR did not discuss the reanalysis of the modified HPSW piping based on the original Construction Code, USAS B31.1-1967 Edition, as a result of the proposed pressure reduction. In its supplement dated March 26, 2019 (ADAMS Accession No. ML19085A385), the licensee stated that it will analyze pipe stress at locations impacted by physical changes associated with the project. Exelon stated that it will reanalyze the HPSW piping in accordance with 1967

(Unit 2) or 1973 (Unit 3, with Addenda through Summer 1973) of the American National Standards Institute (ANSI) B31.1 Code to ensure compliance with ANSI Code requirements. In addition, revisions to calculations for HPSW system piping sections not physically changed by this project will be made to note the new system design pressure and its effect on system piping stresses.

The NRC staff notes that ASME Code, Section XI, IWA-4000, permits the use of the Construction Code for the repair/replacement activities. The NRC staff finds it acceptable that Exelon will analyze the modified HPSW piping in accordance with the Construction Code to demonstrate its structural integrity. The NRC staff further finds that there is reasonable assurance that the proposed change does not affect the compliance of the modified HPSW piping with the Construction Code design requirement, given the reduction of the system pressure.

Exelon stated that there will be no new time critical operator actions for design-basis events required as a result of the change. HPSW system surveillance procedures will be updated to reflect the new design pressure rating. Exelon stated that there are no changes anticipated to the HPSW system radiation monitors, associated control room annunciation, or established procedures for responding to an RHR heat exchanger tube leak. If RHR heat exchanger leakage is detected, the leakage is monitored and included in the Annual Radiological Effluent Release Report, and the heat exchanger is repaired. The existing Peach Bottom leakage detection system will continue to monitor for and alert operators to potential leakage from the RHR system into the modified HPSW system. The NRC staff finds this to be adequate and acceptable.

HPSW Piping/Valve Examination

In a request for additional information dated February 27, 2019 (ADAMS Accession No. ML19058A290), the NRC staff questioned whether Exelon will examine the affected piping and valves prior to installation/replacement to determine the condition of the existing HPSW piping. In its response, Exelon stated that it will perform necessary inspections of existing HPSW piping in areas affected by the installation activities associated with the HPSW project. The areas affected by the modification will include valve, pump, and orifice replacement work. Inspections are required as part of the station's normal maintenance procedures and specifications related to piping replacement and nondestructive examination.

In the above-mentioned request for additional information, the NRC staff asked Exelon to discuss the following: (1) the acceptance examination that will be performed immediately after the hardware modification to demonstrate structural integrity and leaktightness of the modified HPSW piping, (2) whether the hardware modification will be performed based on the provisions of Article IWA-4000 of the ASME Code, Section XI, (3) whether a system leakage test will be performed in accordance with the ASME Code, Section XI, IWA-5000 and IWD-5000, and (4) whether a functionality or operability test will be performed to demonstrate that the proposed pressure reduction will achieve the intended function of the HPSW system.

In its response dated March 26, 2019 (ADAMS Accession No. ML190085A385), Exelon stated that it will perform applicable ASME Code examination and testing as part of the HPSW modification. Exelon stated that the pump and valve replacement meets the provisions of Article IWA-4000, which specify the repair and replacement activities for ASME Code Class 1, 2, and 3, and associated supports. Exelon further stated that the replaced pumps and valves will be subject to an ISI, which includes system leakage, pressure, and visual VT-2 testing, as required in accordance with the ASME Code, Section XI, IWA-5000 and IWD-5000. The pressure test will be performed at the system operating pressure. The acceptance criteria will be based on the new design parameters. The new pump will be tested for flow and pressure, and the valves will be tested for stroke time. These tests will be performed in accordance with the inservice testing requirement to demonstrate operability. Applicable station surveillance tests will be revised, as appropriate, based on the new system design pressure and new valve stroke times. These revised station surveillance tests will be used in the future to demonstrate the system operability to satisfy applicable TS Surveillance Requirements.

Radiological Consequence for Potential Leakage

RG 1.183, Appendix A, Regulatory Position 5, states:

> ESF [engineered safety feature] systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components ... The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs [boiling water reactors] and PWRs [pressurized water reactors].

The intent of Regulatory Position 5 of RG 1.183 is to analyze the radiological consequences for all possible leakage pathways from engineered safety feature systems. The engineered safety feature leakage pathways stated in Regulatory Position 5 are not a complete list of all possible pathways. For Peach Bottom, RHR heat exchanger leakage was not included in the LOCA radiological consequence analysis because the HPSW system was operated at a higher pressure than the RHR system, preventing any outward leakage from the RHR system; accordingly, it was not necessary to evaluate this RHR leakage pathway. However, the proposed change no longer prevents RHR leakage through the heat exchanger interface to the HPSW system, and it allows any RHR leakage to reach the environment via the HPSW system to the discharge canal. In order for the NRC staff to state there is reasonable assurance that the regulatory limits stated in 10 CFR 50.67 continue to be met at Peach Bottom for the LOCA, the RHR leakage through the RHR heat exchanger interface must be reviewed. Therefore, the NRC staff requested the licensee provide a technical evaluation that assesses the impact of the RHR heat exchanger interface leakage during a LOCA analysis and provide a basis for not including the additional radiological dose from this pathway in the license for Peach Bottom, Units 2 and 3, as reflected in the UFSAR design-basis LOCA analysis.

The licensee responded to the above request by letter dated February 15, 2019 (ADAMS Accession No. ML19046A129). The licensee stated that the current design basis, as reflected in the UFSAR for Peach Bottom, is such that: (1) the RHR heat exchangers are designed and were tested to be free from leakage, (2) the design basis does not allow RHR heat exchanger leakage, and (3) the heat exchangers are monitored to identify leakage and ensure the heat exchangers are repaired after leakage is detected. Additionally, the licensee stated that RHR system leakage is required to be minimized, consistent with Peach Bottom procedures, TSs, and the licensing basis. The licensee did not propose any changes to the RHR heat exchanger design requirements with regard to leakage. Furthermore, the licensee provided a scoping analysis of the radiological consequences from RHR heat exchanger leakage to show that there would be a minimal increase in radiological dose to the public, and that the overall radiological dose from a LOCA to the public would remain well below regulatory limits if a LOCA were to occur after an RHR heat exchanger leak had developed.

The NRC staff finds that the licensee's scoping analysis demonstrates that that there is sufficient margin to the regulatory limits stated in 10 CFR 50.67 and 10 CFR Part 50, Appendix A, GDC 19, if a LOCA were to occur after an RHR heat exchanger leak had developed. The licensee did not propose any changes to the RHR heat exchanger design requirements or TS requirements regarding RHR leakage. Based on this, there is reasonable assurance that the licensee's estimates of the radiological consequences from a loss-of-coolant design-basis accident will comply with the requirements of 10 CFR 50.67; 10 CFR Part 50, Appendix A, GDC 19; and the accident-specific dose criteria stated in NUREG-0800, Section 15.0.1, and RG 1.183.

Risk Insights

In the LAR, the licensee stated that the licensee's technical analysis is based on a deterministic evaluation. Therefore, the subject LAR was not a risk-informed request, and a risk evaluation was not required for the purpose of making a regulatory decision.

The NRC staff determined that "special circumstances," as discussed in NUREG-0800, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis; General Guidance," which would have necessitated additional risk information to be provided, did not exist. As such, the NRC staff did not request any additional risk information associated with the review of this LAR.

The licensee provided risk insights related to the proposed change in Section 3.2 of Attachment 5 to the LAR. Because this is not a risk-informed LAR, the staff neither reviewed the licensee's probabilistic risk assessment models to determine their technical acceptability to support this LAR nor relied on the numerical results provided by the licensee. The staff performed an independent assessment using the Peach Bottom Standardized Plant Analysis Risk model. The review of the Peach Bottom Standardized Plant Analysis Risk model and the licensee-provided risk insights supported the traditional engineering conclusions associated with the licensee's proposed compensatory actions. The risk insights did not challenge the engineering conclusions that the proposed changes maintain defense in depth.

As described in the Peach Bottom UFSAR, the HPSW system on each unit has two redundant subsystems, each of which has two HPSW pumps with a common discharge header leading to two RHR heat exchangers in parallel. Either subsystem can provide the required post-accident containment cooling capacity with its two HPSW pumps and two RHR heat exchangers. TS LCO 3.7.1 for each unit requires two HPSW subsystems and the HPSW divisional cross-tie be operable during Modes 1, 2, and 3. Manual initiation of an operable HPSW subsystem is assumed to occur 10 minutes after a design-basis LOCA (DBA-LOCA), with one HPSW pump supplying cooling water to one RHR heat exchanger. One hour after a DBA-LOCA, a second HPSW pump is credited with supplying cooling water to a second RHR heat exchanger for long-term containment cooling. Each of the two credited HPSW pumps must deliver a minimum of 4,500 gpm to each of two RHR heat exchangers for post-LOCA containment cooling. Also, at 1 hour after a DBA-LOCA, one other HPSW pump is credited with delivering 4,500 gpm to one RHR heat exchanger on the non-accident unit for safe shutdown.

After the proposed changes are complete, the HPSW system operating pressures will be lower, but the flow rates will be maintained at or above the existing values. Since there is no change to the design-basis HPSW flow delivery to the RHR heat exchangers and no change to the HPSW supply temperature, the proposed change will have no adverse impact on the RHR heat exchanger thermal performance or containment heat removal capability for normal shutdown operation and design-basis events.

The existing Peach Bottom HPSW system design has its pressure maintained above the RHR system pressure in the RHR heat exchangers while the system is in operation to minimize potential release of radioactivity to the environment via a leak in the RHR heat exchangers. The HPSW design also includes radiation monitors on the HPSW system inlet and outlet piping of the RHR heat exchangers to detect any significant RHR fluid leakage to the HPSW via the RHR heat exchangers. With the proposed change, the HPSW design and operating pressures will be below that of the RHR system. After that change, an RHR heat exchanger tube leak could release radioactivity into the HPSW system water, which is then discharged to the environment. However, the LAR indicates that the existing HPSW system radiation monitoring and control room annunciation capability will not be affected by this design change, and operators would continue to be alerted if such leakage were to occur. Existing administrative controls for response to detection of abnormal radiation conditions will be maintained. Several other boiling water reactor units operate without the service water system pressure being maintained higher than the reactor coolant side of the RHR heat exchangers. The LAR identifies 11 such BWR units, including those at Browns Ferry Nuclear Plant and Limerick Generating Station. This has been found to be acceptable, given the presence of radiation monitors on the service water outlet from their RHR heat exchangers.

The purpose of these TS CTs is to allow a temporary relaxation of the single failure criteria to perform surveillances or necessary, minor maintenance before a reactor shutdown is required. This accommodation of a short but reasonable period is made, understanding that there is an elevated risk level associated with plant transients relative to similar durations of continued steady state operation. The requested changes to extend the TS CTs are one time and of modest additional duration to allow completion of the modifications discussed. With one HPSW subsystem inoperable, the system function is maintained by the operable subsystem. The likelihood of an accident occurring during the CT when designed function redundancy is not present is very low. Plant risk is further controlled by implementation of the compensatory actions described in the LAR and invoked by a note in the proposed TS change verbiage by reference to the LAR:

- Adequate staffing will be maintained onsite to facilitate timely response to unexpected conditions during the extended LCO CTs authorized by the proposed license amendments.

- Equipment will be protected in accordance with the Protected Equipment Program, as described in Procedure OP-AA-108-117. Protected equipment actions taken in accordance with this procedure support the configuration risk management. In addition to the protected opposite HPSW and RHR subsystems required to be operable by the TSs, elective maintenance, discretionary maintenance, and testing on systems that provide support to the protected subsystems (e.g., EDGs) will be suspended during the extended LCO CTs authorized by the proposed license amendments.

- Proper standby alignment of the opposite HPSW and RHR subsystems will be ensured prior to entry into the extended CTs.

- Component testing or maintenance of safety systems in the offsite power systems and important non-safety equipment in the offsite power systems that can increase the likelihood of a plant transient or loss of offsite power, as determined by plant management, will be avoided during the extended LCO CTs authorized by the proposed license amendments.

- Discretionary substation maintenance not being allowed during the extended LCO CTs authorized by the proposed license amendments.
- The high pressure coolant injection pump, reactor core isolation cooling pump, and operable subsystem RHR pumps will not be removed from service for elective maintenance activities during the extended LCO CTs authorized by the proposed license amendments.
- Weather conditions will be monitored, and the HPSW modification work will not be scheduled if severe weather conditions are anticipated.
- The "N+1" diverse and flexible coping strategy (FLEX) pump and FLEX generator will be pre-staged inside the site protected area to allow for more rapid deployment in the event of a loss-of-offsite power with concurrent EDG failures. The use of the "N+1" equipment allows the required FLEX gear to be retained within the protected building, preserving it for an actual FLEX event. This will be controlled in accordance with existing plant procedures.

During the extended LCO CT authorized by the proposed license amendments, operations shift crews will be briefed at the beginning of each shift regarding actions in response to a loss-of-offsite power per applicable plant procedures. Also, all required fire risk management actions will be performed in accordance with site procedures that fulfill the requirements of 10 CFR 50.65(a)(4).

Technical Evaluation Conclusion

The NRC staff finds the proposed changes to the TSs acceptable because the licensee demonstrated through justifiable assumptions and analyses that the regulatory requirements in draft GDC 16, 17, 37, 38, 52, and 70, and 10 CFR 50.36 continue to be met.

The NRC staff's review finds that the radiological consequences for the DBA-LOCA design-basis accident are not affected by the proposed modification to the HPSW system and the proposed TS changes and that the regulatory limits stated in 10 CFR 50.67; 10 CFR Part 50, Appendix A, GDC 19; and the accident-specific criteria stated in NUREG-0800, Section 15.0.1, and RG 1.183 continue to be met at Peach Bottom. Therefore, the licensee's proposed modification to the HPSW system and proposed TS changes are acceptable from a radiological dose perspective.

On the basis of its review, the NRC staff concludes that the proposed piping modification is acceptable because the NRC staff finds that: (1) Exelon will replace HPSW piping components in accordance with the repair/replacement activities of the ASME Code, Section XI, IWA-4000; (2) Exelon will reanalyze the HPSW piping in accordance with the Construction Code of ANSI B 31.1-1967; (3) Exelon has demonstrated that the reduced system pressure will not affect the functionality and operability of the HPSW system; (4) Exelon has demonstrated that structural integrity of the replaced HPSW piping will be maintained; and (5) the requested changes to extend the TS CTs are one time and of modest additional duration to allow completion of the modifications.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments on August 5, 2019. 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. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (83 FR 55566). 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: K. Bucholtz
J. Tsao
J. Bettle
S. Jones

Date: August 28, 2019

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – ISSUANCE OF AMENDMENT NOS. 327 AND 330 RE: REDUCE HIGH PRESSURE SERVICE WATER SYSTEM DESIGN PRESSURE AND TEMPORARILY EXTEND COMPLETION TIMES (EPID L-2018-LLA-0265) DATED AUGUST 28, 2019

DISTRIBUTION:
Public
PM File Copy
RidsRgn1MailCenter Resource
RidsACRS_MailCTR Resource
RidsNrrDssScpb Resource
RidsNrrDssSrxb Resource
RidsNrrDssStsb Resource
RidsNrrDeEmib Resource
RidsNrrDmlrMphb Resource
RidsDraArcb Resource
RidsNrrDorlLpl1 Resource
RidsNrrLACSmith Resource
RidsNrrLALRonewicz Resource
RidsNrrPMPeachBottom Resource
SCumblidge,NRR
KBucholtz, NRR
JTsao, NRR
JBettle, NRR
SJones, NRR

ADAMS Accession No.: ML19182A006 *by memorandum **by e-mail

OFFICE	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/LA	NRR/DE/EMIB/BC*	NRR/DMLR/MPHB*
NAME	JTobin	CSmith	SBailey	SCumblidge
DATE	07/29/2019	07/23/2019	07/30/2019	04/25/2019
OFFICE	NRR/DSS/STSB/BC**	NRR/DSS/SCPB/BC**	NRR/DRA/ARCB/BC*	NRR/DSS/SRXB/BC*
NAME	PSnyder	SAnderson	KHsueh	JWhitman
DATE	08/06/2019	06/20/2019	05/07/2019	04/10/2019
OFFICE	OGC – NLO**	NRR/DORL/LPL1/BC	NRR/DORL/LPL1/PM	
NAME	KGamin	JDanna	JTobin	
DATE	08/12/2019	09/03/2019	08/28/2019	

OFFICIAL RECORD COPY