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

July 5, 2016

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

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – ISSUANCE OF AMENDMENTS RE: SURVEILLANCE REQUIREMENTS FOR HIGH PRESSURE COOLANT INJECTION SYSTEM AND REACTOR CORE ISOLATION COOLING SYSTEM (CAC NOS. MF6774 AND MF6775)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendments Nos. 308 and 312 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3. The amendments consist of changes to the technical specifications in response to your application dated October 2, 2015, as supplemented by letter dated March 23, 2016.

The amendments (1) revise the allowable test pressure band in the technical specification surveillance requirements (SRs) for the pump flow testing of the high pressure coolant injection system and the reactor core isolation system; (2) revise the surveillance frequency requirements for verifying the sodium pentaborate enrichment of the standby liquid control system; and (3) delete SRs associated with verifying the manual transfer capability of the normal and alternate power supplies for certain motor-operated valves associated with the suppression pool spray and drywell spray sub-systems of the residual heat removal system.

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. 308 to Renewed DPR-44
2. Amendment No. 312 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. 308
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 October 2, 2015, as supplemented by letter dated March 23, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 308, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

for

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

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

Date of Issuance: July 5, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 308

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 area of change.

Remove	Insert
3.1-23	3.1-23
3.5-6	3.5-6
3.5-13	3.5-13
3.6-30	3.6-30
3.6-30b	3.6-30b

(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. 308, 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.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.7	Deleted	
SR 3.1.7.8	Verify each pump develops a flow rate ≥ 49.1 gpm at a discharge pressure ≥ 1275 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.9	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program.
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 92.0 atom percent B-10.	Prior to addition to SLC tank

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.8	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1053 and ≥ 915 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.1.9	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 175 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.1.10	------------------NOTE------------------ Vessel injection/spray may be excluded. -- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.3	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1053 psig and ≥ 915 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.4	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 175 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify each RHR suppression pool spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.2.4.2	Verify each suppression pool spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.2.4.3	Deleted	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.5.1	Verify each RHR drywell spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.2.5.2	Verify each drywell spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.2.5.3	Deleted	

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. 312
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 October 2, 2015, as supplemented by letter dated March 23, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 312, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

For

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

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

Date of Issuance: July 5, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 312

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 area of change.

Remove	Insert
3.1-23	3.1-23
3.5-6	3.5-6
3.5-13	3.5-13
3.6-30	3.6-30
3.6-30b	3.6-30b

(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) <u>Maximum Power Level</u>

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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 312, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) <u>Physical Protection</u>

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

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

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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.7	Deleted	
SR 3.1.7.8	Verify each pump develops a flow rate $\geq$ 49.1 gpm at a discharge pressure $\geq$ 1275 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.9	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program.
SR 3.1.7.10	Verify sodium pentaborate enrichment is $\geq$ 92.0 atom percent B-10.	Prior to addition to SLC tank

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.8	------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1053 and ≥ 915 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.1.9	------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 175 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.1.10	------------------NOTE------------------- Vessel injection/spray may be excluded. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.3	------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1053 psig and ≥ 915 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.4	------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 175 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify each RHR suppression pool spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.2.4.2	Verify each suppression pool spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.2.4.3	Deleted	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.5.1	Verify each RHR drywell spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.2.5.2	Verify each drywell spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program.
SR 3.6.2.5.3	Deleted	

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 308 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-44 AND

AMENDMENT NO. 312 TO RENEWED FACILITY OPERATING LICENSE NO. 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 October 2, 2015, as supplemented by letter dated March 23, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML15275A265 and ML16083A394, respectively), Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3.

The amendments would (1) revise the allowable test pressure band in the technical specification (TS) surveillance requirements (SRs) for pump flow testing of the high pressure coolant injection (HPCI) system and the reactor core isolation (RCIC) system; (2) revise the surveillance frequency requirements for verifying the sodium pentaborate enrichment of the standby liquid control (SLC) system; and (3) delete SRs associated with verifying the manual transfer capability of the normal and alternate power supplies for certain motor-operated valves associated with the suppression pool spray and drywell spray sub-systems of the residual heat removal (RHR) system.

The supplement dated March 23, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 8, 2015 (80 FR 76320).

2.0 REGULATORY EVALUATION

2.1 System Descriptions

2.1.1 General Description

PBAPS, Units 2 and 3, are boiling-water reactor (BWR) plants of the BWR/4 design with Mark-I type pressure suppression containment. As described in Section 5.1.2 of the Updated Final Safety Analysis Report (UFSAR), the primary containment encloses the reactor vessel, the reactor coolant recirculation system, and other branch connections of the reactor coolant system. The major elements of the primary containment are the drywell, the pressure suppression chamber (or wetwell) that stores a large volume of water (suppression pool), the connecting vent pipe system between the drywell and the wetwell, isolation valves, vacuum breakers, containment cooling systems, and other service equipment.

2.1.2 HPCI and RCIC Systems

The emergency core cooling system (ECCS) is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss-of-coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS consists of the HPCI system, the core spray (CS) system, the low pressure coolant injection mode of the RHR system, and the automatic depressurization system. The suppression pool provides the required source of water for the ECCS.

The HPCI system is designed to provide core cooling in the event of a small-break LOCA that does not depressurize the reactor quickly enough to permit timely operation of the low pressure ECCS. The HPCI system accomplishes this function by injecting coolant makeup water into the reactor pressure vessel (RPV) with a steam turbine-driven pump unit.

The RCIC system is not part of the ECCS; however, it has similar functions. The RCIC system is designed to operate either automatically or manually following RPV isolation, accompanied by a loss-of-coolant flow from the feedwater system to provide core cooling and control of the RPV water level. The RCIC system uses a steam turbine-driven pump unit to perform its design functions.

2.1.3 SLC System

The standby liquid control (SLC) system provides backup capability for reactivity control independent of the control rod system. The SLC system functions by injecting a boron solution into the reactor to effect shutdown.

The SLC system consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the RPV. The SLC system is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods.

2.1.4 RHR Suppression Pool Spray and Drywell Spray

Following a design-basis accident (DBA), the RHR suppression pool spray system removes heat from the suppression chamber airspace. The suppression pool is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the RPV through safety/relief valves. The heat addition to the suppression pool results in increased steam in the suppression chamber, which increases primary containment pressure. Some means must be provided to remove heat from the suppression chamber so that the pressure and temperature inside primary containment remain within analyzed design limits. This function is provided by two redundant RHR suppression pool spray subsystems.

Drywell spray is a mode of the RHR system that may be initiated under post-accident conditions to reduce the temperature and pressure of the primary containment atmosphere. The drywell spray function is credited in design-basis analyses to limit peak drywell temperature following a steam line break inside of the drywell and may be used to mitigate other LOCAs inside of the drywell. This function is provided by two redundant drywell spray subsystems.

2.2 Proposed TS Changes

2.2.1 Change Test Pressure Band for HPCI and RCIC SRs 3.5.1.8 and 3.5.3.3

The amendments would revise the allowable test pressure band in SR 3.5.1.8 for the HPCI system pump flow test and in SR 3.5.3.3 for the RCIC system pump flow test. SRs 3.5.1.8 and 3.5.3.3 currently require the pump flow tests to be performed with reactor pressure between 940 pounds per square inch gauge (psig) and 1,053 psig. The amendments would change the lower pressure limit value from 940 psig to 915 psig. As discussed in the application dated October 2, 2015, the licensee stated that this change would reduce challenges to the control of reactor pressure and reactivity when performing the surveillances during plant startup operations. The specific changes are shown below.

SR 3.5.1.8 currently reads as follows:

> Verify, with reactor pressure ≤ 1053 and ≥ 940 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm [gallons per minute] against a system head corresponding to reactor pressure.

SR 3.5.18 would be revised to read as follows:

> Verify, with reactor pressure ≤ 1053 and ≥ 915 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.

SR 3.5.3.3 currently reads as follows:

> Verify, with reactor pressure ≤ 1053 psig and ≥ 940 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.

SR 3.5.3.3 would be revised to read as follows:

> Verify, with reactor pressure ≤ 1053 psig and ≥ 915 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.

2.2.2 Change Frequency for SLC System SR 3.1.7.10

SR 3.1.7.10 provides requirements to verify the sodium pentaborate enrichment of the SLC system. The frequency for performance of this SR currently reads as follows:

> In accordance with the Surveillance Frequency Control Program
>
> AND
>
> Once within 8 hours after addition to SLC tank

The amendments would revise the frequency for performance of SR 3.1.7.10 to read as follows:

> Prior to addition to SLC tank

As discussed in the application dated October 2, 2015, the licensee stated that this change would delete the unnecessary requirement for controlling the frequency within the Surveillance Frequency Control Program (SFCP) and would provide a frequency consistent with the Standard Technical Specifications (STSs).

2.2.3 Delete RHR Suppression Pool Spray SR 3.6.2.4.3 and RHR Drywell Spray SR 3.6.2.5.3

The amendments would delete RHR suppression pool spray SR 3.6.2.4.3 and RHR drywell spray SR 3.6.2.5.3. Both of these SRs currently read as follows:

> Verify manual transfer capability of power supply for the RHR motor-operated flow control valve and the RHR cross-tie motor-operated valve from the normal source to the alternate source.

As discussed in the application dated October 2, 2015, the licensee stated that the alternate power supply capability is not necessary for the suppression pool spray and drywell spray sub-systems of the RHR system to perform their design function. As such, the SRs are not needed to demonstrate operability of the associated limiting conditions for operation (LCOs).

2.3 Regulatory Requirements and Guidance

The regulatory requirements and guidance that the NRC staff considered in its review of this LAR are described below.

2.3.1 Technical Specification Requirements

In Title 10 of the *Code of Federal Regulations* (10 CFR) Section 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)(3), SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

2.3.2 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 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 include 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 has made changes to the facility over the life of the plant that may have invoked the final GDC. The extent to which the final GDC have been invoked can be found in specific sections of the UFSAR and in other plant-specific design and licensing basis documentation.

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

- Draft GDC 44, "Emergency Core Cooling Systems Capability (Category A)," which requires, in part, that at least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function

and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary.

2.3.3 Other Regulatory Requirements

Paragraph (c)(4) in 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram events for light-water-cooled nuclear power plants," requires, in part, that the SLC system be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control.

2.3.4 Guidance Documents

The NRC staff used NUREG-1433, Revision 4, "Standard Technical Specifications - General Electric BWR/4 Plants" (ADAMS Accession Nos. ML12104A192 and ML12104A193), for guidance on TS format and content.

3.0 TECHNICAL EVALUATION

3.1 Change Test Pressure Band for HPCI and RCIC SRs 3.5.1.8 and 3.5.3.3

The PBAPS TSs for the HPCI and RCIC system pump flow surveillance tests, SR 3.5.1.8 and SR 3.5.3.3, respectively, currently require the tests to be performed with a reactor pressure ≥ 940 psig and ≤ 1,053 psig to confirm that the required pump flow can be achieved. The amendments would change the lower pressure limit value from 940 psig to 915 psig.

By Amendment Nos. 293 and 296 (PBAPS, Units 2 and 3, respectively) dated August 25, 2014 (ADAMS Accession No. ML14133A046), the NRC approved a 12.4 percent extended power uprate (EPU) that authorized an increase in the maximum thermal power level from 3,514 megawatts thermal (MWt) to the current licensed thermal power level of 3,951 MWt. The EPU did not change the maximum allowable reactor steam dome pressure of 1,053 psig, as specified in TS 3.4.10. In order to achieve the EPU power level, an increase in feedwater flow to the reactor and an increase in steam flow from the reactor to the main turbine occurs. This increased steam flow rate causes an increased pressure drop between the reactor steam dome and the main turbine control valves and requires a lower pressure at the main turbine control valve inlets. As such, as a result of the EPU, the licensee stated that the electrohydraulic control (EHC) pressure regulator setpoint at the turbine control valves has been lowered from 940 psig to 915 psig for 100 percent steady-state power operations.

As discussed in the licensee's application dated October 2, 2015, during plant startup and power ascension testing, it is desirable to set the pressure regulator setpoint at a value that will support 100 percent steady state power operations without the need to adjust the pressure regulator. Unnecessary adjustments to the pressure regulator can result in reactivity and reactor pressure perturbations. The licensee stated that a pressure regulator setpoint of 915 psig will control pressure in the reactor steam dome at approximately 1,030 to 1,035 psig at 100 percent steady-state power operations. This provides margin to the TS 3.4.10 reactor steam dome pressure limit of 1,053 psig. During low power operations, there is much less steam flow to the

turbine. The licensee stated that a pressure regulator setpoint of 915 psig results in the reactor steam dome pressure value being close to 915 psig during low power operations.

The licensee stated that maintaining the current lower test pressure of 940 psig, as currently specified in SR 3.5.1.8 and SR 3.5.3.3, would result in an unnecessary perturbation of reactivity and reactor pressure, since the pressure regulator setting would need to be increased to 940 psig to perform the SR 3.5.1.8 and SR 3.5.3.3 testing, and then once again lowered to 915 psig following testing, to facilitate higher Mode 1 power operations. The licensee further stated that changing the lower test pressure value from 940 to 915 psig in SR 3.5.1.8 for the HPCI system and SR 3.5.3.3 for the RCIC system can reduce the potential for a plant transient.

The HPCI pump is designed to provide a flow of 5,000 gallons per minute (gpm) over a reactor pressure range of 150 to 1,170 psig. The TS SR flow tests for the HPCI system are performed at two different pressure ranges such that system capability to provide the rated flow of 5,000 gpm is tested at both the lower and upper portions of the operating range of the system. SR 3.5.1.8 tests at a higher portion of the pressure range (i.e., currently 940 to 1,053 psig), and SR 3.5.1.9 tests at a lower portion of the pressure range (less than or equal to 175 psig).

The RCIC pump is designed to provide a flow of 600 gpm over a reactor pressure range of 150 to 1,170 psig. The TS SR flow tests for the RCIC system are performed at two different pressure ranges such that system capability to provide the rated flow of 600 gpm is tested at both the lower and upper portions of the operating range of the system. SR 3.5.3.3 tests at a higher portion of the pressure range (i.e., currently 940 to 1,053 psig), and SR 3.5.3.4 tests at a lower portion of the pressure range (less than or equal to 175 psig).

The LAR does not propose to change the required HPCI flow rate of ≥ 5,000 gpm currently specified in SR 3.5.1.8 or the required RCIC flow rate of ≥ 600 gpm currently specified in SR 3.5.3.3. Only the lower pressure value of 940 psig in SR 3.5.1.8 and SR 3.5.3.3 would be changed to 915 psig.

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. SR 3.5.1.8 is performed to demonstrate that the HPCI pump provides the required flow rate of at least 5,000 gpm in order to demonstrate operability of the HPCI system in accordance with the associated LCO 3.5.1. Similarly, SR 3.5.3.3 is performed to demonstrate that the RCIC pump provides the required flow rate of at least 600 gpm in order to demonstrate operability of the RCIC system in accordance with the associated LCO 3.5.1.

As noted above, both the HPCI and RCIC pumps are designed to provide their required flow rates at a maximum reactor pressure of 1,170 psig. The maximum design pressure for HPCI and RCIC corresponds approximately to the lowest group of safety relief valve (SRV) setpoints (including drift) as discussed in Section 3.4.1 of the NRC staff's safety evaluation (SE) for Amendment Nos. 290 and 293 (PBAPS, Units 2 and 3, respectively) dated May 5, 2014 (ADAMS Accession No. ML14079A102). Specifically, as shown in SR 3.4.3.1, the lowest set of SRV setpoints has a lift setting of 1135 ± 34.1 psig. However, maximum reactor steam dome pressure is limited to ≤ 1,053 psig in accordance with TS 3.4.10. As such, the maximum reactor steam dome pressure of 1,053 psig in TS 3.4.10 corresponds to the upper end of the pressure

test range specified in SR 3.5.1.8 and SR 3.5.3.3. The LAR does not propose to change the 1,053 psig value in these SRs.

With respect to the current lower pressure value of 940 psig in SR 3.5.1.8 and SR 3.5.3.3, that value was established in Amendment Nos. 212 and 217 (PBAPS, Units 2 and 3, respectively) dated January 11, 1996 (ADAMS Accession No. ML011500051). As discussed in the NRC staff's SE for those amendments, the lower pressure value of 940 psig is consistent with the minimum EHC pressure setpoint at which reactor power can be increased without the need to adjust the EHC pressure setpoint during operation in Mode 1. As noted above, as a result of the recent EPU, the EHC pressure regulator setpoint at the turbine control valves has been lowered from 940 psig to 915 psig for 100 percent steady-state power operations. As such, the proposed change to SR 3.5.1.8 and SR 3.5.3.3 would revise the lower pressure value consistent with the design and licensing basis that was in place following the 1996 amendments and prior to the EPU issued in 2014 (i.e., the proposed lower value of 915 psig is consistent with the minimum EHC pressure setpoint at which reactor power can be increased without the need to adjust the EHC pressure setpoint during operation in Mode 1).

The NRC staff finds that the proposed change to SR 3.5.1.8 and SR 3.5.3.3 provides an appropriate range of values to verify the operation of the HPCI and RCIC systems at the upper end of the reactor pressure operating range. As such, the staff further finds that the change continues to meet the intent of 10 CFR 50.36(c)(3) with respect to establishing SRs that provide assurance that the associated LCOs will be met. Based on these findings, the staff concludes that changing the lower pressure limit value from 940 psig to 915 psig in SR 3.5.1.8 and SR 3.5.3.3 is acceptable.

3.2 Change Frequency for SLC System SR 3.1.7.10

SR 3.1.7.10 requires verification of the sodium pentaborate enrichment of the SLC tank. The current frequency for performance of SR 3.1.7.10 was established as part of the EPU amendment. The current frequency reads as follows:

> In accordance with the Surveillance Frequency Control Program
>
> AND
>
> Once within 8 hours after addition to SLC tank

The SFCP is described in TS 5.5.14. The test intervals, and changes to the test intervals for SRs included in the SFCP, are controlled by the licensee in accordance with the program. Prior to the EPU amendment, the frequency for performance of SR 3.1.7.10 was, "Once within 8 hours after addition to the SLC tank." The EPU amendment added the wording "In accordance with the Surveillance Frequency Program" to the frequency requirements for SR 3.1.7.10. However, the licensee's application dated October 2, 2015, indicated that the licensee never established a new frequency in the SFCP. As such, the frequency for performance of this SR was the same as the pre-EPU frequency of, "Once within 8 hours after addition to the SLC tank." Thus, in the application, the licensee has requested to remove the wording "In accordance with the Surveillance Frequency Program" from the frequency requirements for SR 3.1.7.10.

Additionally, the licensee has proposed to change the frequency for performance of SR 3.1.7.10 to, "Prior to addition to SLC tank." This change would provide wording identical to the STSs for BWR/4 plants (i.e., NUREG-1433, Revision 4).

Enriched sodium pentaborate solution is made by mixing granular enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual Boron-10 (B-10) enrichment are performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used.

The NRC staff finds that performance of the SR prior to addition to the SLC tank provides reasonable assurance that the SLC tank enrichment will be maintained within limits consistent with demonstrating operability of the SLC system in accordance with LCO 3.1.7. As such, the staff further finds that the change continues to meet the intent of 10 CFR 50.36(c)(3) with respect to establishing SRs that provide assurance that the associated LCOs will be met. In addition, the staff finds that the proposed change will continue to ensure SLC tank B-10 enrichment consistent with the requirements in 10 CFR 50.62(c)(4). Based on these findings, the staff concludes that the proposed change to SR 3.1.7.10 is acceptable.

3.3 Delete RHR Suppression Pool Spray SR 3.6.2.4.3 and RHR Drywell Spray SR 3.6.2.5.3

3.3.1 Functions of RHR System Containment Cooling Operating Modes

As described in Enclosure 9c to Attachment 9 to the licensee's EPU application dated September 28, 2012 (ADAMS Package Accession No. ML122860201), the RHR system is composed of two loops designated Division I and Division II. Each division includes two RHR pumps and two heat exchangers (HXs). The A and C pumps and HXs are in Division I, and the B and D pumps and HXs are in Division II. Each division's two pumps, two HXs, and associated piping are referred to as a loop. The RHR system has two loops for all modes of operation. Each loop is in a separate area of the reactor building to minimize the possibility of a single physical event causing the loss of the entire system.

As described in UFSAR Section 4.8.6.2, the RHR system provides safety-related means to cool the containment when operating in its suppression pool cooling (SPC) or drywell spray (DWS) and suppression pool spray (SPS) modes. The safety function of the SPC mode is to remove decay and sensible heat discharged to the suppression pool and maintain its temperature and the containment pressure below an acceptable limit during and following a DBA or a special event. In the DWS and SPS modes, the water pumped through the RHR system HXs is diverted to spray headers in the drywell (DWS mode) and above the suppression pool (SPS mode). The DWS condenses steam that may exist in the drywell, thereby lowering containment pressure. The spray water collects in the bottom of the drywell until the level rises to the level of the drywell vent lines, where it overflows and drains into the suppression pool. In the SPS mode, approximately 5 percent of the total flow may be directed to the suppression chamber spray ring to cool any non-condensable gases collected in the free volume above the suppression pool.

3.3.2 Background

In the EPU license amendment, the licensee eliminated the containment accident pressure (CAP) credit for calculating the available net positive suction head of the core spray and RHR

system pumps that draw water from the suppression pool during a DBA or special events. For this purpose, the licensee performed the following modifications in both units: (a) installed a safety-related 10-inch diameter cross-tie line with a normally closed motor-operated isolation valve between the discharge lines of RHR pumps A and C and, separately, between the discharge lines of RHR pumps B and D; (b) replaced the existing flow restriction orifices upstream of the RHR HXs B and C with new motor-operated flow control valves; (c) replaced the existing control valve upstream of the RHR HXs A and D with new motor-operated flow control valves; and (d) installed alternate power supply capability to RHR HX cross-tie valves, the RHR flow control valves, and the high pressure service water (HPSW) system valves to assure containment cooling in the event of a single failure along with loss of offsite power (LOOP). This modification included the addition of control room capability for the manual transfer of the power supply from a safety-related normal source to a safety-related alternate source for the RHR system and the HPSW valves. The EPU license amendment added SR 3.6.2.3.3 for the SPC mode of RHR system, SR 3.6.2.4.3 for the SPS mode of RHR system, and SR 3.6.2.5.3 for the DWS mode of RHR system. The purpose of these SRs is as follows:

- SR 3.6.2.3.3 tests the capability of transferring power to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves from the alternate power supply in the SPC mode of the RHR system.
- SR 3.6.2.4.3 tests the capability of transferring power to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves from the alternate power supply in the SPS mode of the RHR system.
- SR 3.6.2.5.3 tests the capability of transferring power to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves from the alternate power supply in the DWS mode of the RHR system.

Subsequent to the NRC approval of the EPU licensee amendment, the licensee re-evaluated the failure modes and effects analysis (FMEA) results and determined that the failure of the power supply transfer to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves from the alternate power supply will not affect the RHR to perform its safety-related containment cooling without using CAP. The FMEA results concluded that the capability of the testing of the power supply transfer to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves from the alternate power supply is only needed for the SPC mode of the RHR system so that SR 3.6.2.3.3 is only required to satisfy 10 CFR 50.36 (c)(3) criteria. Therefore, the licensee proposed to delete SRs 3.6.2.4.3 and 3.6.2.5.3.

In the supplement dated March 23, 2016, the licensee described the relationships between the SPS and DWS modes of RHR system, the emergency electrical power supplies, and the need for the alternate power supplies to compensate for a postulated emergency electrical power supply failure. The licensee stated that the RHR system is designed with two subsystems (i.e., loops), with the A subsystem comprised of the A and C RHR trains powered by the E1 and E3 emergency diesel generators (EDGs) and the B subsystem compromised of the B and D RHR trains powered by the E2 and E4 EDGs. The A subsystem SPS and DWS injection valves are powered by the E3 EDG. The corresponding B subsystem valves are powered by the E4 EDG. In the event of a LOOP and a single active failure of the E3 or E4 EDG, the SPS and DWS valves in the subsystem with the failed EDG will not have emergency electrical power.

The RHR alternate power supply was installed to ensure that the RHR HX cross-tie, RHR flow control, and the HPSW outlet valves are provided with the capability of receiving power from either their normal source or from an alternate safety-related power supply.

3.3.3 Justification for Deletion of SR 3.6.2.4.3 and SR 3.6.2.5.3

The licensee stated that while the use of the alternate power supply to the RHR HX cross-tie, RHR flow control, and the HPSW outlet valves may be used for the operation of the HX cross-tie in the SPS and DWS modes of RHR, the need for this capability would involve more than a single failure. The alternate power supply capability for the specific RHR valves is not relied upon for the design function of the SPS and DWS modes of RHR. The accident analyses do not rely on the use of the alternate power supply for the SPS and DWS modes of RHR, assuming any single failure.

In the supplement dated March 23, 2016, the licensee justified the availability of containment cooling without the use of CAP for a single failure of one EDG, concurrent with the failure of power transfer capability from the alternate power supply to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves in the SPS and DWS modes of RHR, for several scenarios. The license stated that the bounding accident analysis for a large-break LOCA is a recirculation line break (RSLB) DBA LOCA in one unit with a LOOP and safe shutdown of the second unit. The licensee further stated that the bounding accident analysis for the peak primary containment suppression pool, drywell airspace, and drywell shell temperature is a small steam line break (SSLB) LOCA in one unit with a LOOP and safe shutdown of the second unit. The licensee stated both events must be accompanied with the limiting single failure of the loss of one EDG. A summary of these scenarios is provided below.

RSLB DBA LOCA with LOOP, Safe Shutdown of Second Unit, and Loss of E3 or E4 EDG

In this scenario, the SPS and DWS modes of RHR are not available for containment cooling because the failed EDG powers their spray injection valves. Containment cooling is performed in the coolant injection cooling (CIC) mode by one RHR pump with flow through one HX during the first hour, and the two RHR cross-tied HXs after the first hour, using the valves powered by the alternate power supply. In the CIC mode, the RHR system provides both SPC and reactor cooling. In this mode, water is drawn from the suppression pool by the RHR pump, cooled through the HX(s), injected into the RPV, and returned to the suppression pool through the break. In the SPC mode, the RHR system provides direct SPC, and the CS system provides reactor cooling.

The NRC staff finds that for this scenario, the licensee's evaluation justifies that the RHR SPS and DWS modes are not relied upon to perform the containment cooling function when the alternate power supply is needed (i.e., after 1 hour).

RSLB DBA LOCA with LOOP, Safe Shutdown of Second Unit, and Loss of E1 or E2 EDG

In this scenario, the DWS and SPS modes of RHR are available in the presence of normal power. However, these modes would not be available when the HX cross-tie is needed in the presence of a LOOP concurrent with the failure of power transfer capability from the alternate power supply to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves.

Therefore, these modes of RHR are not relied upon for the containment cooling function. The safety basis for the containment function is by the CIC mode of the RHR without the HX cross-tie in operation during the first hour, and with cross-tie in operation after the first hour, with the cross-tie being always available in the CIC and SPC modes of the RHR system.

The NRC finds that for this scenario, the licensee's evaluation justifies that the RHR SPS and DWS modes are not relied upon to perform the containment cooling function, coincident with the use of the RHR HX cross-tie operation. Therefore, the alternate power supply is not relied on for the RHR SPS and DWS modes of operation for this scenario.

SSLB LOCA with LOOP, Safe Shutdown of the Second Unit, and Loss of One EDG

In this scenario, the CS pumps are not needed. Two EDGs would be available to power the RHR pumps, including the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves to mitigate the LOCA in the accident unit. Two RHR pumps and two HXs would be available in the RHR SPS and DWS modes and will not rely on the RHR HX cross-tie for mitigation of the accident. In case of failure of the power transfer capability of the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves to the alternate power supply, the SPS and DWS modes of RHR without the cross-tie capability will be able to mitigate the accident. Adequate emergency power would be available for the safe shutdown of the non-accident unit.

The NRC finds that for this scenario, the licensee's evaluation justifies that the alternate power supply to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves is not needed for the SPS and DWS modes of the RHR system.

Other Potential Design-Basis Scenarios

In a request for additional information (RAI) designated as SCVB-RAI-5, the NRC staff cited several examples in Attachment 6 to the licensee's EPU application, which discussed crediting of the DWS and SPS modes of the RHR system for the containment cooling function following a LOCA. The staff questioned whether there were any cases in the design basis where the DWS or SPS modes of RHR rely on the alternate power supply to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves. The licensee provided its response in its supplement dated March 23, 2016.

The NRC staff has reviewed the information provided by the licensee and finds that the licensee has demonstrated that the examples cited by the staff did not represent cases where the alternate power supply to the RHR HX cross-tie, RHR flow control, and RHR HX HPSW outlet valves is needed for the DWS or SPS modes of the RHR system.

Containment Atmosphere Cleanup Function

In SCVB-RAI-2, the licensee was requested to explain whether the SPS and DWS modes of the RHR system are credited for containment atmosphere cleanup consistent with 10 CFR, Part 50, Appendix A, GDC-41, and if so, whether the alternate power supply was needed to perform this function. This GDC requires systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released

into the environment following postulated accidents. In the supplement dated March 23, 2016, the licensee stated that the SPS and DWS modes of RHR are not credited for fission product cleanup. The licensee stated that the systems credited for this function include primary containment and its isolation valves, the SLC system, secondary containment, and the standby gas treatment system.

The NRC staff reviewed the licensee's response to SCVB-RAI-2 and accepts that the SPS and DWS modes of RHR system are not credited for the containment atmosphere fission product cleanup function.

Conclusion

Based on the above, the NRC staff finds that the alternate power supply capability is not necessary for the SPS and DWS modes of the RHR system to perform their design functions. Therefore, the NRC staff concludes that the test of the alternate power supply transfer capability does not meet the 10 CFR 50.36(c)(3) criteria to be included as an SR for the SPS and DWS modes of the RHR system. As such, deletion of SR 3.6.2.4.3 and SR 3.6.2.5.3 is acceptable.

3.4 Technical Evaluation Conclusion

Based on the discussion in SE Sections 3.1 through 3.3 above, the NRC staff concludes that the proposed amendments are acceptable.

Attachment 3 to the licensee's application dated March 24, 2016, provided revised TS Bases pages to be implemented with the associated TS changes. These pages were provided for information only and will be revised in accordance with the TS Bases Control Program discussed in PBAPS TS 5.5.10.

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 change SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (80 FR 76320). 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: R. Ennis
A. Sallman
F. Forsaty

Date: July 5, 2016

July 5, 2016

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

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – ISSUANCE OF AMENDMENTS RE: SURVEILLANCE REQUIREMENTS FOR HIGH PRESSURE COOLANT INJECTION SYSTEM AND REACTOR CORE ISOLATION COOLING SYSTEM (CAC NOS. MF6774 AND MF6775)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendments Nos. 308 and 312 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 in response to your application dated October 2, 2015, as supplemented by letter dated March 23, 2016.

The amendments (1) revise the allowable test pressure band in the technical specification surveillance requirements (SRs) for the pump flow testing of the high pressure coolant injection system and the reactor core isolation system; (2) revise the surveillance frequency requirements for verifying the sodium pentaborate enrichment of the standby liquid control system; and (3) delete SRs associated with verifying the manual transfer capability of the normal and alternate power supplies for certain motor-operated valves associated with the suppression pool spray and drywell spray sub-systems of the residual heat removal system.

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. 308 to Renewed DPR-44
2. Amendment No. 312 to Renewed DPR-56
3. Safety Evaluation

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ADAMS Accession No.: ML16159A148

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DATE	6/10/2016	6/10/2016	6/15/2016	6/16/2016
OFFICE	OGC	DORL/LPL1-2/BC	DORL/LPL1-2/PM	
NAME	JLindell	DBroaddus (JPoole for)	REnnis	
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