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

February 26, 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. 323 AND 326 TO REVISE TECHNICAL SPECIFICATIONS TO ALLOW CONTINUED OPERATION WITH TWO SAFETY RELIEF VALVES/SAFETY VALVES OUT OF SERVICE (EPID L-2018-LLA-0151)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 323 and 326 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3, respectively. These amendments are in response to your application dated May 30, 2018, as supplemented by letter dated December 6, 2018 (Agencywide Documents Access and Management System Accession Nos. ML18150A387 and ML18340A185, respectively).

The amendments revise the Peach Bottom, Unit 2 and 3, Technical Specifications to allow continued operation with two safety relief valves/safety valves out of service and to increase the reactor coolant system pressure safety limit. Specifically, the amendments revise Technical Specification Safety Limit 2.1.2 and Limiting Condition for Operation 3.4.3 for both Units 2 and 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. 323 to DPR-44
2. Amendment No. 326 to 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. 323
Renewed License No. DPR-44

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company) and PSEG Nuclear LLC (the licensees), dated May 30, 2018, as supplemented by letter dated December 6, 2018, 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. 323, 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 the date of its issuance and shall be implemented within 60 days.

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 Technical Specifications
and Facility Operating License

Date of Issuance: February 26, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 323

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of Renewed Facility Operating License No. DPR-44 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
Page 3	Page 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
2.0-1	2.0-1
3.4-8	3.4-8

(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. 323, 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.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

THERMAL POWER shall be ≤ 22.6% RTP.

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

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

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1340 psig.

2.2 SL Violations

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety Relief Valves (SRVs) and Safety Valves (SVs)

LCO 3.4.3 The safety function of 12 valves (any combination of SRVs and and SVs) shall be OPERABLE.

---------------------------NOTE---------------------------
The safety function of 12 valves (any combination of SRVs and SVs) are required to be OPERABLE $\leq$ 3358 MWt during operating cycle 22.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs or SVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

UNITED STATES
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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. 326
Renewed License No. DPR-56

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company) and PSEG Nuclear LLC (the licensees), dated May 30, 2018, as supplemented by letter dated December 6, 2018, 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. 326, 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 the date of its issuance and shall be implemented within 60 days.

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 Technical Specifications
and Facility Operating License

Date of Issuance: February 26, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 326

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

RENEWED FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following page of Renewed Facility Operating License No. DPR-56 and Appendix A 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
Page 3	Page 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
2.0-1	2.0-1
3.4-8	3.4-8

(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. 326, 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.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

THERMAL POWER shall be ≤ 22.6% RTP.

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

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

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1340 psig.

2.2 SL Violations

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety Relief Valves (SRVs) and Safety Valves (SVs)

LCO 3.4.3 The safety function of 12 valves (any combination of SRVs and SVs) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs or SVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 323 AND 326

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-44 AND DPR-56

EXELON GENERATION COMPANY, LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By letter dated May 30, 2018 (Reference 1), as supplemented by letter dated December 6, 2018 (Reference 2), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) to allow continued operation with two safety relief valves (SRVs)/safety valves (SVs) out of service and to increase the reactor coolant system (RCS) pressure safety limit. The license amendments would revise Technical Specification (TS) Safety Limit 2.1.2 and Limiting Condition for Operation (LCO) 3.4.3 for the Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom).

The supplement dated December 6, 2018, 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 determined as published in the *Federal Register* on November 6, 2018 (83 FR 55564).

2.0 REGULATORY EVALUATION

General Design Criteria

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 Updated Final Safety Analysis Report (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 (hereinafter referred to as the "draft GDC"). Appendix H of 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" (hereinafter referred to as the "final GDC"). Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the NRC Staff Requirements Memorandum, SECY-92-223, "Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (Reference 3), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. At the time of the promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the final GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the final GDC were formally adopted was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission.

The licensee for Peach Bottom 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 9, "Reactor Coolant Pressure Boundary (Category A)," which requires that the reactor coolant pressure boundary (RCPB) be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

- Draft GDC 14, "Core Protection Systems (Category B)," which requires that core protection systems, together with associated equipment, be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

- Draft GDC 15, "Engineered Safety Features Protection Systems (Category B)," which requires that protection systems be provided for sensing accident situations and initiating the operation of necessary engineered safety features (ESFs).

- Draft GDC 28, "Reactivity Hot Shutdown Capability (Category A)," which requires that at least two of the reactivity control systems provided be independently capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

- Draft GDC 29, "Reactivity Shutdown Capability (Category A)," which requires, in part, that at least one of the reactivity control systems provided be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits.

- Draft GDC 30, Reactivity Holddown Capability (Category B)," which requires that at least one of the reactivity control systems provided be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

- Draft GDC 37, "Engineered Safety Features Basis for Design (Category A)," which requires, in part, that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems.

- Draft GDC 41, "Engineered Safety Features Performance Capability (Category A)," which requires, in part, that ESFs such as emergency core cooling and containment heat removal systems provide the required safety function, assuming a failure of a single active component.

- Draft GDC 42, "Engineered Safety Features Components Capability (Category A)," which requires that ESFs be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident (LOCA).

- Final GDC 15, "Reactor coolant system design," which requires the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- Final GDC 31, "Fracture prevention of reactor coolant pressure boundary," which requires the RCPB be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

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

Technical Specification Requirements

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

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

As discussed in 10 CFR 50.36(c)(3), surveillance requirements 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.

As discussed in 10 CFR 50.36(c)(5), administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

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

Other Regulatory Requirements

The regulations in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," in part, establish standards for the calculation of emergency core cooling system (ECCS) accident performance and acceptance criteria for that calculated performance.

The regulations in 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," in part, establish standards for the calculation of ATWS performance and acceptance criteria for the calculated performance:

- Each boiling water reactor must have an alternate rod injection system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

- Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gallons per minute (gpm) of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel.

- Each boiling water reactor must have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS. ATWS is defined as an anticipated operational occurrence (AOO) followed by the failure of the reactor trip portion of the protection system.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Change

TS Section 2.1.2, "Reactor Coolant System Pressure SL [Safety Limit]," currently states:

> Reactor steam dome pressure shall be ≤ 1325 psig.

The licensee's proposed revision would state:

> Reactor steam dome pressure shall be ≤ 1340 psig.

TS Section 3.4.3, "Safety Relief Valves (SRVs) and Safety Valves (SVs)," LCO 3.4.3, currently states:

> The safety function of 13 valves (any combination of SRVs and SVs) shall be OPERABLE.

The licensee's proposed revision would state:

> The safety function of 12 valves (any combination of SRVs and SVs) shall be OPERABLE.

3.2 System Description

Peach Bottom is a boiling water reactor (BWR) of General Electric BWR/4 design, with a Mark I containment. The nuclear boiler system transports the steam generated in the reactor pressure vessel through the primary containment by means of a piping system (consisting of four 26-inch main steam lines with two pneumatically operated, globe type isolation valves in each steam line) from the reactor pressure vessel nozzles to the outboard main steam isolation valves (MSIVs). Between the reactor pressure vessel and the MSIVs, three SVs and 11 dual function SRVs are mounted on the steam lines which, in conjunction with reactor scram, assist in limiting peak pressure in the primary system during plant transient conditions. The SVs and SRVs of the nuclear boiler system are designed to meet the requirements for reactor vessel overpressure protection to conform to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, Article 9. The design pressure of the reactor vessel and RCPB is 1,250 pounds per square inch gauge (psig). The acceptance limit for pressurization events is the ASME Code allowable peak pressure of 1,375 psig (110 percent of design value). The nuclear system pressure relief system is designed with 11 SRVs with opening setpoints of 1,135 psig, 1,145 psig, and 1,155 psig, and 3 SVs with opening setpoints of 1,260 psig.

The SRVs are Target Rock three-stage pilot operated safety/relief valves. The SVs are Dresser spring loaded SVs. The SRVs and SVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The SRVs can actuate by either of two modes: the safety mode or the depressurization mode. In the safety mode, the pilot disc opens when steam pressure at the valve inlet expands the bellows to the extent that the hydraulic seating force on the pilot disc is reduced to zero. Opening of the pilot stage allows a pressure differential to develop across the second stage disc, which opens the second stage disc, thus venting the chamber over the main valve piston. This causes a pressure differential across the main valve piston, which opens the main valve. The SVs are spring loaded valves that actuate when steam pressure at the inlet overcomes the spring force holding the valve disc closed.

Each of the 11 SRVs discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The three SVs discharge steam directly to the drywell. In the depressurization mode, each SRV is opened by a pneumatic actuator that opens the second stage disc. The main valve then opens as described above for the safety mode. The depressurization mode is initiated either manually by the operator or automatically by the automatic depressurization system (ADS). Unlike the safety mode, the depressurization mode does not rely on the pilot stage and is independent of the bellows. The depressurization mode provides a method for depressurization of the reactor coolant pressure boundary. All 11 of the SRVs function in the safety mode and have the capability to operate in the depressurization mode by manual actuation. Five of the SRVs are allocated to the ADS.

The safety objective of the pressure relief system is to prevent overpressurization of the nuclear system. This protects the RCPB from failure, which could result in the uncontrolled release of fission products. In addition, the automatic depressurization feature of the pressure relief

system acts in conjunction with the ECCS for reflooding the core. This protects the reactor fuel cladding from failure due to overheating.

3.3 NRC Staff Evaluation

As described in the LAR, the basis for the licensee to propose the TS change is a reevaluation of the transient pressure analysis at the current licensed thermal power authorized by the measurement uncertainty recapture (MUR) uprate amendments (Reference 4). This analysis is provided in Attachment 6, "GE Hitachi Nuclear Energy 004N6240-NP, 'Peach Bottom Units 2 & 3 Two Safety Relief Valves Out-of-Service Evaluation,'" Revision 1 (Non-Proprietary Version), to the LAR.

The licensee identified the safety analyses that are potentially affected by the proposed TS change. The staff reviewed the LAR and other license documents and finds that the licensee has identified the appropriate, potentially affected safety analyses. The affected analysis areas as identified in the LAR include ASME overpressure protection, ATWS, ECCS-LOCA performance, and high pressure system performance. The staff reviewed the licensee's evaluation on these areas below with the regulatory requirements identified in Section 2.0 of this safety evaluation.

3.3.1 ASME Overpressure Analysis

Overpressure protection for the RCPB during power operation is provided by SRVs, SVs, and the RPS. Draft GDC 9 requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime. Final GDC 15 requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. Final GDC 31 requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating fracture is minimized.

Currently, the limiting overpressure AOO event for Peach Bottom is the main steam isolation valve closure with scram on high flux (MSIVF). The case of MSIVF is analyzed during every cycle-specific reload and was reevaluated at MUR conditions to assure that the ASME Code allowable value for peak vessel pressure was not breached. By complying with the ASME Code, the above-mentioned GDC, especially final GDC 15, are directly met. The MSIVF case analysis conservatively assumed that the MSIV position scram fails and the event terminates on a high neutron flux scram signal. The closure of all MSIVs causes a rapid pressure increase in the reactor vessel. The pressure increase is mitigated by the actuation of the SRVs and SVs. The Peach Bottom, Units 2 and 3, maximum extended load line limit analysis plus (MELLLA+) amendments dated March 21, 2016 (Reference 5), and the reload analysis for the current operating cycles for Peach Bottom, Units 2 and 3, confirmed that the MSIVF event at the current licensed thermal power (CLTP) of 4,016 megawatts thermal (MWt) remains the limiting overpressure event with 13 of 14 operable SRVs/SVs. The overpressure analyses assume that the out-of-service relief valve is an SRV with the lowest pressure setpoint. This assumption is conservative since this minimizes the initial pressure relief capacity and results in the highest peak pressure value for the overpressure analyses. Hence, the analyses are bounding for any SRV or SV out of service (OOS). This analysis forms the current overpressure analysis.

To evaluate the impact of the proposed two SRV/SV OOS, the licensee reanalyzed the overpressure analysis for the MSIVF event by using the most recent reload licensing analysis

inputs for Peach Bottom, Units 2 and 3, at CLTP conditions with GNF2 fuel, with the SRVs/SVs configuration at a ±3 percent tolerance setting for each valve, and with two SRVs at the lowest pressure setpoint out of service (two SRV/SV OOS). This selection of SRVs minimizes the initial pressure relief capacity and bounds any combination of SRVs and/or SVs out of service for overpressure analysis. This analysis was performed using the TRACG AOO methodology (Reference 6 and Reference 7) at CLTP of 4,016 MWt, with consideration for the MELLLA+ operating domain. The MSIVF event was analyzed at both minimum (85 percent of rated) and maximum (110 percent of rated) core flow, which are bounding conditions for reactor vessel overpressure calculations. Both Peach Bottom units have reactor cores with only GNF2 fuel assemblies. The TRACG statistical pressure adder was calculated for GNF2 fuel for use in the ASME Code overpressure analysis per the TRACG AOO methodology.

The reanalyzed results for two SRV/SV OOS show that the peak steam dome pressure of 1,326 and 1,327 psig for Units 2 and 3, respectively, will exceed the current dome pressure SL of 1,325 psig for both Peach Bottom units. The licensee claimed that the corresponding calculated peak vessel pressure (1,355 and 1,356 psig for Units 2 and 3, respectively) is still below the ASME overpressure limit of 1,375 psig by about 20 pounds per square inch (psi). Since the dome pressure SL is selected to prevent the integrity of the RCS from being endangered (Peach Bottom TS Base 2.1.2), the licensee proposed that a rise of dome pressure SL by 15 psi to 1,340 psig will still meet the ASME overpressure limit of 1,375 psig with 5 psi margin. With the proposed new dome pressure SL, the peak steam dome pressure with two SRV/SV OOS will then be below the SL with a margin of about 15 psi. The NRC staff requested additional information by e-mail dated November 8, 2018 (Reference 8), to clarify and ensure that the proposed new dome SL will prevent the integrity of the RCS from being endangered.

In its December 6, 2018, response to the NRC's request for additional information, the licensee verified and confirmed that the peak vessel pressure values reported in Tables 1 and 2 of Attachment 1 to the LAR represent the pressure in the lowest RCS volume of the TRACG-AOO model. The elevation of the vertical midpoint of this volume is 0.65 meters (2.1 feet (ft)) above the bottom of the vessel. The static elevation head of this additional 2.1 feet of elevation above the bottom of the RCS is less than 1 psi. The licensee also confirmed that the TRACG statistical pressure adder was applied to the peak vessel pressure as reported in Tables 1 and 2 of Attachment 1 to the LAR.

Since the final GDC 15 requires the RCS be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation (including AOOs), the NRC staff raised a concern that the peak vessel pressure might not remain below the ASME limit of 1,375 psig (with margin) when the steam dome pressure approaches the proposed safety limit (1,340 psig) on the steam dome pressure (Reference 8). The licensee replied in the same request for additional information response that, for such a case (i.e., if the steam dome pressure approaches 1,340 psig), the peak vessel pressure will be approximately 6 psi less than the ASME Code pressure limit (Reference 2). The peak vessel pressure, as determined, was based on a finding from the past (e.g., Peach Bottom, Unit 2, Cycle 22, for the limiting increased core flow case and Peach Bottom, Unit 3, Cycle 21, for the limiting increased core flow case) and current MSIVF results analyzed for Peach Bottom that a maximum pressure differential of about 30 psi exists between peak vessel and steam dome. This maximum pressure differential has not been changed from cycle to cycle due to the unchanged maximum allowable core flow and fuel bundle hydraulic characteristics. The licensee further stated that its reload analysis process and the associated cycle-specific Supplemental Reload Licensing Report will evaluate and report both the steam dome and peak

vessel pressure results to verify/validate that neither the TS SL nor the ASME Code limit is exceeded.

The staff found the licensee's response acceptable for the justification of the limiting ASME overpressure event MSIVF results under the new operating condition (i.e., two SRV/SV OOS) and the proposed new SL on steam dome pressure because the provided justification (i.e., past analysis experiences and physical setup for the transients) is reasonable and conforms to the ASME Code. Therefore, the staff concludes that the proposed change, i.e., allowing two SRVs/SVs to be OOS and increasing the reactor pressure SL, will continue to meet the draft GDC 9, final GDC 15, and final GDC 31.

3.3.2 ATWS Overpressure Analysis

An ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in draft GDC 14 and 15. Regulatory requirements related to ATWS events are specified in 10 CFR 50.62. The NRC staff review includes confirming that the peak reactor vessel bottom pressure will be less than the ASME Service Level C limit of 1,500 psig during an ATWS overpressure event as protected by the SRVs/SVs and systems required in accordance with 10 CFR 50.62 (e.g., alternate rod injection system, SLCS).

The plant-specific ATWS overpressure analysis is performed using the approved overpressure methodology (Reference 6 and Reference 9) with the reactor operating at the MUR bounding high power level (4,018 MWt) and limiting MELLLA+ core flow conditions. Two events were considered: main steam isolation valve closure (MSIVC) and pressure regulator failure open (PRFO). Each of these events has the potential to yield the maximum vessel overpressure result. In the MSIVC event, the MSIVs on all four steam lines close simultaneously, while the normal, direct scram on full MSIVC fails. In addition, scram on high neutron flux and high vessel pressure are assumed to fail. In the PRFO event, the pressure regulator failure produces the maximum steam flow demand. Reactor pressure drops and the MSIVs close on a low steam line pressure signal. Similarly, scram on full MSIV closure, high flux, and high pressure all fail. The ultimate shutdown of the plant is accomplished through the actuation of the SLCS. The initial reduction in power occurs as a result of the ATWS high dome pressure recirculation pump trip. After the ATWS recirculation pump trip, and following the opening of the SRVs/SVs, the event is terminated for overpressure considerations.

To evaluate the impact of the proposed two SRV/SV OOS on ATWS, the licensee simulated ATWS with two SRV/SV OOS and the SRV/SV opening setpoints at +3 percent of the nominal setpoint. The resulting peak vessel pressure values and comparison between the one and two SRV/SV OOS conditions for the MSIVC and pressure regulator failure open events are provided in Attachment 6 (Section 2.2.2, first paragraph) to the LAR. The ATWS overpressure analysis results with two SRV/SV OOS are still below the ASME Code Service Level C limit of 1,500 psig (emergency condition) so that the ATWS overpressure requirement as stated in 10 CFR 50.62 continues to be met.

The staff finds the ATWS overpressure assessment and associated response with two SRV/SV OOS is acceptable because it meets the applicable criterion for ATWS regulatory requirements as required by draft GDC 14 and 15 and 10 CFR 50.62.

3.3.3 <u>Other Assessments</u>

The staff reviewed the other systems and analyses that are potentially affected by the proposed TS change, including ECCS-LOCA performance and high pressure systems performance. The staff's evaluation of the licensee's assessments on these systems and analyses is provided below.

3.3.3.1 ECCS-LOCA Performance Assessment

The regulations in 10 CFR 50.46 establish standards for the calculation of ECCS performance and acceptance criteria for that calculated performance. Draft GDC 42 requires that ESFs be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a LOCA. Draft GDC 41 requires that the emergency core cooling and containment heat removal systems provide the required safety function, assuming a failure of a single active component.

The Peach Bottom ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The licensee evaluated the impact of SRV/SV OOS on the ECCS-LOCA performance via the consideration of the following two break size categories.

For large break LOCA cases, depressurization of the vessel occurs due to mass and energy release from the break. Therefore, further relief through the SRVs is not required to arrive at post-accident conditions whereby ECCS assets can be delivered to recover the core and stop the temperature excursion on the cladding.

For small break LOCA cases, the SRVs are relied upon for depressurization through the ADS. The SRVs can be actuated by alternate means, mechanical or pneumatic, so a valve declared out of service for one actuation may still be available for the alternate actuation. ADS availability is stipulated by TS Section 3.5.1, "ECCS−Operating," with assurance that the five SRVs supporting ADS would remain available despite changes to allow two SRV/SV OOS (mechanically, for overpressure protection) under the provisions of TS Section 3.4.3, "SRVs and SVs." With this confirmation, and noting that the five SRVs available for ADS function conform to the basis of the most recent ECCS-LOCA analysis (Reference 5) for limiting small break cases, it is concluded that there would be no effect on the ECCS-LOCA analysis results with two SRV/SV OOS.

The staff finds that the licensee's LOCA-ECCS performance evaluation with respect to the use of SRVs is acceptable because the impact of the use of SRVs on the reactor fuel and core responses under LOCA conditions has been evaluated reasonably and consistently with the TSs. The staff finds the evaluation acceptable since the proposed two SRV/SV OOS will not have impact on the existing LOCA-ECCS analysis results that are in compliance with 10 CFR 50.46 and draft GDC 41 and 42. Therefore, the staff concludes that the proposed TS change is acceptable for the LOCA-ECCS because the LOCA-ECCS related regulatory requirements in 10 CFR 50.46 and draft GDC 41 and 42 will continue to be met.

3.3.3.2 High Pressure System Performance Assessment

The high pressure systems include the high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), standby liquid control (SLC), and control rod drive (CRD) systems. For HPCI, draft GDC 37 and 41 require that a system provide abundant emergency core cooling so

that fuel and cladding damage that would interfere with the emergency core cooling function will be prevented. For RCIC, draft GDC 37 requires that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems.

For the CRD and SLC systems, draft GDC 28 requires that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Draft GDC 29 and 30 require that at least one of the reactivity control systems be capable of making and holding the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits. The regulation in 10 CFR 50.62(c)(4) requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provide a set level of reactivity control.

The staff reviewed the licensee's evaluation and found that the most significant potential effect of two SRV/SV OOS on the high pressure systems' operations is the maximum reactor pressure at which they are required to deliver water to the reactor. Details for the evaluation are provided below.

The HPCI system is designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel. In this event, the HPCI system maintains reactor water level and helps depressurize the reactor vessel. The adequacy of the HPCI system is demonstrated in the ECCS performance discussion in Section 3.3.3.1 of this safety evaluation. The HPCI system also serves as a backup to the RCIC system. The HPCI backup function for an RCIC system failure is unaffected because the injection rate of the HPCI system is significantly greater than the RCIC system. The HPCI system is required to provide injection into the reactor pressure vessel at the lowest group of SRV setpoints (including the +3 percent tolerance drift), provided that there are at least two functional SRVs in the lowest group. Because there are four SRVs in the lowest group, taking two SRV/SV OOS in this group still leaves two SRVs. Therefore, the HPCI system injection capability is not affected by an additional SRV/SV OOS so that draft GDC 37 and 41 continue to be met.

The RCIC system is required to maintain sufficient water inventory in the reactor to permit adequate core cooling following a reactor vessel isolation event accompanied by loss of flow from the feedwater system. The system design injection rate must be sufficient to comply with the system limiting criteria to maintain the reactor water level above the top of active fuel. The RCIC system is designed to pump water into the reactor vessel over a wide range of operating pressures. The RCIC system is required to provide injection into the reactor pressure vessel at the lowest group of SRV setpoints (including the +3 percent tolerance drift), provided that there are at least two functional SRVs in the lowest group. Because there are four SRVs in the lowest group, taking two SRV/SV OOS in this group still leaves two SRVs. Therefore, the Peach Bottom RCIC system injection capability is adequate to support two SRV/SV OOS. The RCIC system with two SRV/SV OOS continues to meet draft GDC 37.

The SLCS is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that some or all of the control rods cannot be inserted. This manually operated system pumps a highly enriched sodium pentaborate solution into the reactor vessel to provide neutron absorption and achieve a subcritical reactor condition. The SLCS is designed to inject over a wide range of reactor operating pressures. The original SLCS design criteria for

the maximum system operating pressure was based on the SRVs operating in the relief mode at the upper analytical setpoint limit. The SLCS is not dependent upon any other SRV operating modes. This criterion has been generally replaced by using plant-specific ATWS transient pressure data occurring during the time the SLCS is analyzed to be in operation in consideration of NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin," dated August 10, 2001 (Reference 10). ATWS specific pressure data generally exceeds the original injection pressure. This injection pressure was calculated for both extended power uprate and MELLLA+ and is within the capability of the SLCS. For the case of two SRV/SV OOS, the ATWS results show that the SRVs are cycling with a maximum relief flow of less than rated steam flow. This relief flow is well within the capacity of the remaining nine in-service SRVs, and it is concluded that the peak reactor vessel pressure calculated for SLCS injection will not be exceeded with an additional SRV/SV OOS. Therefore, the Peach Bottom SLCS is adequate to support two SRV/SV OOS and continues to meet draft GDC 28, 29, and 30 and 10 CFR 50.62(c)(4).

The CRD system is designed to shut down the reactor by inserting control rods. The CRD mechanism is also part of the reactor pressure boundary. For the two SRV/SV OOS condition, the licensee provided its evaluation in LAR Attachment 4, Section 3.3. This evaluation concludes that the effect of the two SRV/SV OOS on control rod injection times is bounded by the current analysis. The evaluation also concludes that, for CRD integrity, the CRD mechanism has been analyzed (LAR Attachment 4, Section 3.3) to a value that bounds the peak vessel pressures as shown in Tables 1 and 2 of Attachment 1 to the LAR. Therefore, the CRD system is adequate to support two SRV/SV OOS and continues to meet draft GDC 28, 29, and 30.

The NRC staff reviewed the licensee's evaluation and determined that the impact of allowing two SRV/SV OOS and the changed reactor coolant system pressure SL will not impair the ability of the relevant safety systems to maintain their design functions as required by draft GDC 9, 14, 15, 28, 29, 30, 37, 41, and 42; final GDC 15 and 31; 10 CFR 50.36, 10 CFR 50.46; and 10 CFR 50.62, with the proposed operating condition. Therefore, the NRC staff concludes that the proposed TS change is acceptable.

3.4 Technical Summary

The NRC staff finds the proposed TS amendments acceptable because the licensee demonstrated through justifiable assumptions and analyses that the regulatory requirements in draft GDC 9, 14, 15, 28, 29, 30, 37, 41, and 42; final GDC 15 and 31; 10 CFR 50.36; 10 CFR 50.46; and 10 CFR 50.62, continue to be met.

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 January 15, 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 55564). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from Exelon Generation Company, LLC to U.S. NRC, "Peach Bottom Atomic Power Station, Units 2 and 3, License Amendment Request – Revise Technical Specifications Section to Allow Two Safety Relief Valves/Safety Valves to be Out-of-Service with Increased Reactor Pressure Safety Limit," dated May 30, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18150A387).

2. Letter from Exelon Generation Company, LLC to U.S. NRC, "Peach Bottom Atomic Power Station, Units 2 and 3, Response to Request for Additional Information, License Amendment Request – Revise Technical Specifications to Allow Two Safety Relief Valves/Safety Valves to be Out-of-Service with Increased Reactor Pressure Safety Limit," dated December 6, 2018 (ADAMS Accession No. ML18340A185).

3. U.S. Nuclear Regulatory Commission Staff Requirements Memorandum, SECY-92-223, "Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736).

4. Letter from U.S. NRC to Exelon Generating Company, LLC, "Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments Re: Measurement Uncertainty Recapture Power Update (CAC Nos. MF9289 and MF9290; EPID L-2017-LLS-0001)," dated November 15, 2017 (ADAMS Accession No. ML17286A013).

5. Letter from U.S. NRC to Exelon Generation Company, LLC, "Peach Bottom Atomic Station, Units 2 and 3, Issuance of Amendments Re: Maximum Extended Load Line Limit Analysis Plus," dated March 21, 2016 (ADAMS Accession No. ML16034A372).

6. GE Hitachi Nuclear Energy, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," NEDE-32906P Supplement 3-A, Revision 1, dated April 2010.

7. GE Nuclear Energy, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," NEDE-32906P-A, Revision 3, dated September 2006.

8. U.S. NRC e-mail to Exelon Generation Company, LLC, "Peach Bottom Units 2 and 3 – Request for Additional Information (public) – LAR to Allow 2 SRV/SVs OOS at High Pressure (EPID L-2018-LLA-0151)," dated November 8, 2018 (ADAMS Accession No. ML18312A407).

9. GE Nuclear Energy, "TRACG Application for Anticipated Transient Without Scram Overpressure Transient Analyses," NEDE-32906P Supplement 1-A, dated November 2003.

10. U.S. NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin," dated August 10, 2001 (ADAMS Accession No. ML012210146).

Principal Contributors: S. Peng
I. Tseng

Date: February 26, 2019

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – ISSUANCE OF AMENDMENT NOS. 323 AND 326 TO REVISE TECHNICAL SPECIFICATIONS TO ALLOW CONTINUED OPERATION WITH TWO SAFETY RELIEF VALVES/SAFETY VALVES OUT OF SERVICE (EPID L-2018-LLA-0151) DATED FEBRUARY 26, 2019

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