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

April 18, 2018

Mr. Peter P. Sena, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 323 AND 304 RE: RELOCATION OF REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE TABLES (EPID L-2017-LLA-0313)

Dear Mr. Sena:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 323 and 304 to Renewed Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2, respectively. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated September 27, 2017.

The amendments relocate the Reactor Coolant System Pressure Isolation Valve Table from the TSs to the Technical Requirements Manual. The amendments also remove references to the table and move all notes and leakage acceptance criteria from the table to the TS surveillance requirements.

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

Sincerely,

for

James Kim, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 323 to Renewed DPR-70
2. Amendment No. 304 to Renewed DPR-75
3. Safety Evaluation

cc: Listserv

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

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 323
Renewed License No. DPR-70

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

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC (the licensees) dated September 27, 2017, 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 set forth in 10 CFR Chapter I;

   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-70 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 323, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications, and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James Danna

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

Attachment:
Changes to Renewed Facility Operating License and Technical Specifications

Date of Issuance: April 18, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 323

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following page of Renewed Facility Operating License No. DPR-70 with the attached revised page as indicated. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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| Page 3 | Page 3 |

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove | Insert |
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| 3/4 4-16a | 3/4 4-16a |
| 3/4 4-16b | 3/4 4-16b |
| 3/4 4-16c | 3/4 4-16c |

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at a steady state reactor core power level not in excess of 3459 megawatts (one hundred percent of rated core power).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 323, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications, and the Environmental Protection Plan.

(3) Deleted Per Amendment 22, 11-20-79

(4) Less than Four Loop Operation

PSEG Nuclear LLC shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this renewed license) with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensees and approval for less than four loop operation at power levels above P-7 has been granted by the Commission by Amendment of this renewed license.

(5) PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety

REACTOR COOLANT SYSTEM

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor Coolant System Pressure Isolation Valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the specified limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.3 Each Reactor Coolant System Pressure Isolation Valve shall be demonstrated OPERABLE pursuant to the INSERVICE TESTING PROGRAM, except that in lieu of any leakage testing required by the INSERVICE TESTING PROGRAM, each valve shall be demonstrated OPERABLE by verifying leakage to be ≤ 5.0 gpm each valve[a][b]:

a. In accordance with the Surveillance Frequency Control Program.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

d. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valve. For all other systems testing will be done once per refueling.

The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or 4.

SURVEILLANCE REQUIREMENTS (Continued)

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(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable. However, for initial tests, or tests following valve repair or replacement, leakage rates less than or equal to 5.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum differential test pressure shall not be less than 150 psid.

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

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 304
Renewed License No. DPR-75

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

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC (the licensees) dated September 27, 2017, 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 set forth in 10 CFR Chapter I;

   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-75 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 304, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James Danna

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

Attachment:
Changes to Renewed Facility Operating License and Technical Specifications

Date of Issuance: April 18, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 304

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following page of Renewed Facility Operating License No. DPR-75 with the attached revised page as indicated. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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| Page 3 | Page 3 |

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

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| 3/4 4-18 | 3/4 4-18 |
| 3/4 4-19 | 3/4 4-19 |

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at steady state reactor core power levels not in excess of 3459 megawatts (thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 304, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. NOT USED

f. 1 GPM leakage at a Reactor Coolant System pressure of 2230 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, and primary-to-secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor in accordance with the Surveillance Frequency Control Program.

b. Monitoring the containment sump inventory in accordance with the Surveillance Frequency Control Program.

c*. Verifying primary-to-secondary leakage is ≤ 150 gallons per day through any one steam generator in accordance with the Surveillance Frequency Control Program during steady state operation,

d*. Performance of a Reactor Coolant System water inventory balance** in accordance with the Surveillance Frequency Control Program. The water inventory balance shall be performed with the plant at steady state conditions. The provisions of specification 4.0.4 are not applicable for entry into Mode 4, and

e. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve shall be demonstrated OPERABLE pursuant to the INSERVICE TESTING PROGRAM, except that in lieu of any leakage testing required by the INSERVICE TESTING PROGRAM, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. In accordance with the Surveillance Frequency Control Program.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

d. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valve. For all other systems testing will be done once per refueling.

The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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* Not required to be completed until 12 hours after establishment of steady state operation.

** Not applicable to primary-to-secondary leakage.

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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 304 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

## 1.0 INTRODUCTION

By letter dated September 27, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17270A076), PSEG Nuclear LLC (PSEG or the licensee) submitted a request for changes to the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2, Technical Specifications (TSs). The requested changes would relocate the Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Table from the TSs to the Technical Requirements Manual (TRM). The request would also remove references to the table and move all notes and leakage acceptance criteria from the table to the TS surveillance requirements (SRs).

## 2.0 REGULATORY EVALUATION

### 2.1 Background

The TS requirements for the Salem RCS PIVs are specified in Salem, Unit No. 1, TS 3/4.4.6.3, "Primary Coolant System Pressure Isolation Valves," and Salem, Unit No. 2, TS 3/4.4.7.2, "Operational Leakage." The function of the RCS PIVs is to separate the high pressure RCS from an attached low pressure system in order to prevent overpressure failure of the low pressure system, which could lead to a loss-of-coolant accident (LOCA). The Salem RCS PIV TS requirements limit the allowable PIV leakage to amounts that do not compromise safety in Modes 1-4. At lower RCS pressures (Modes 5 and 6), there is a reduced potential for leakage.

In accordance with SRs 4.4.6.3 (Unit No. 1) and 4.4.7.2 (Unit No. 2), a PIV:

> ...shall be demonstrated OPERABLE pursuant to the INSERVICE TESTING PROGRAM, except that in lieu of any leakage testing required by the

INSERVICE TESTING PROGRAM, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limits:

a. In accordance with the Surveillance Frequency Control Program.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

d. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves [...] the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valves. For all other systems testing will be done once per refueling.

The Salem TSs include a list of PIVs that the leakage limits are applicable to.

### 2.2 Regulatory Requirements and Guidance

The U.S. Nuclear Regulatory Commission (NRC or the Commission) staff identified the following regulatory requirements and guidance as being applicable to the license amendment request.

#### 2.2.1 General Design Criteria

Salem was designed in accordance with the Atomic Industrial Forum General Design Criteria and the licensee's understanding of the intent of the Atomic Energy Commission proposed General Design Criteria published in 1967. The licensee performed a comparison of the Salem, Unit Nos. 1 and 2, plant design and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC), dated July 7, 1971. This comparison was documented in the Salem Updated Final Safety Analysis Report (UFSAR), Section 3.1.3, which concludes, in part, that the Salem plant design conforms with the intent of the GDC, with the exceptions noted, including GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment." In its letter dated September 27, 2017, PSEG identified the following GDCs in Appendix A to 10 CFR Part 50 as applicable to the proposed amendments:

Criterion 14 – Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

Criterion 54 – Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy,

> reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

The licensee also identified 10 CFR Part 50, Appendix A, GDC 55, as being applicable to the proposed amendments. However, as noted above, the Salem design does not conform with the intent of GDC 55. Salem UFSAR Section 6.2.4, discusses the valves that do not conform with GDC 55. In part, Section 6.2.4.1 states:

> Per acceptance methods of general Design Criteria 55 and 56 and ANS N2711976 56.2 the two barriers [between the atmosphere outside the containment and the containment atmosphere] may consist of: (a) two closed piping systems or vessels, one inside and one outside the containment, (b) two automatic isolation valves, one inside and one outside the containment, (c) an automatic isolation valve inside the containment and a closed system outside the containment, (d) an automatic isolation valve outside the containment and close system inside the containment, or (e) an automatic isolation valve outside containment and a closed system outside the containment.

Also, UFSAR Section 6.2.4.1, states that "[a] check valve on an incoming line or a normally closed valve is considered an automatic valve." This differs from GDC 55, which states that a, "simple check valve may not be used as the automatic isolation valve outside containment." Thus, the NRC staff does not consider GDC 55 to be fully applicable to the proposed amendment.

### 2.2.2 Technical Specification Requirements

The NRC's regulatory requirements related to the content of TSs are specified in 10 CFR 50.36, "Technical specifications." 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) limiting conditions for operations (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 stated in 10 CFR 50.36(c)(2)(i), LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met. The LCO action requirements establish those remedial actions that must be taken when the requirements of an LCO are not met.

As stated 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.

On July 22, 1993 (58 FR 39132), the Commission published the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (Final Policy Statement), which discussed the criteria to determine which items are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to

10 CFR 50.36 (60 FR 36953; July 19, 1995). Specifically, 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

As discussed in the *Federal Register* notice for the final rule dated July 19, 1995 (60 FR 36955):

> LCOs that do not meet any of the criteria, and their associated actions and surveillance requirements, may be proposed for relocation from the technical specifications to licensee-controlled documents, such as the FSAR [Final Safety Analysis Report]. The criteria may be applied to either standard or custom technical specifications.

As discussed in the Final Policy Statement (58 FR 39138):

> When licensees submit amendment requests based on this Policy Statement, they should identify the location of and controls for the technical and administrative requirements of the relocated requirements. The NRC staff will carefully review these submittals to ensure the accountability and the acceptability of controls for each relocated requirement. Many of the requirements will be relocated to the FSAR and will be enforceable through 10 CFR 50.59. Other requirements will be relocated to more appropriate documents (e.g., Security Plan, QA Plan) and controlled by the applicable regulatory requirements. The adequacy of controls for relocated requirements which do not fit in the above categories will be reviewed and approved by the NRC staff on a case-by-case basis to determine, among other things, whether an enforceable control method will need to be established.

As discussed in the licensee's letter dated September 27, 2017, the proposed amendments would relocate certain TS requirements (as discussed in Section 2.3 of this safety evaluation) to the Salem TRM. The TRM is a PSEG-controlled document that has been developed to contain requirements relocated from the TSs. The TRM is described in Section 13.5, "Plant Procedures," of the Salem UFSAR. Specifically, UFSAR Section 13.5.4 reads as follows:

> Technical Requirements Manual (TRM)
>
> The Technical Requirements Manual (TRM) contains technical requirements and/or supporting information (e.g., tables and component lists) which were once contained in the SGS [Salem Generating Station] Technical Specifications (TS) (i.e., Appendix A of the SGS Facility Operating License). Removal of the TS and information is approved by the NRC through individual TS amendments. The TRM is intended to provide operational guidance and requirements for various plant conditions, actions, and testing similar to TS, however, these requirements are in accordance with licensing commitments. These changes add the TRM into the scope of procedures to be processed through the Station Qualified Reviewer (SQR) process and reviewed by PORC [Plant Operations Review Committee]. Future changes to the relocated requirements and supporting information are processed in accordance with section 17.2 of the UFSAR, and are subject to a 10 CFR 50.59 Review. All non-editorial changes are reviewed by PORC.
>
> The TRM is comprised of an index, the individual specification and bases. The manual is intended to provide a single location for the relocated TS items as a convenience for operations and other station personnel. The individual sections of the TRM contain the relocated licensing commitments which are subject to the provisions of 10 CFR 50.59 described above, and are controlled in accordance with the applicable established procedure process.

Nuclear Energy Institute (NEI) guidance document NEI 98-03, Revision 1, "Guidelines for Updating Final Safety Analysis Reports" (ADAMS Accession No. ML003779028), lists the following methods of controlling the TRM on page 7 of Appendix A:

> The TRM or other licensee controlled document is explicitly "incorporated by reference" into the UFSAR. Under this approach, the referenced document is subject to the change control requirements of 10 CFR 50.59 and the updated/reporting requirements of 10 CFR 50.71(e), e.g., periodic submittal of change pages, etc.
>
> The TRM or other licensee controlled document is treated in a manner consistent with the procedures fully or partially described in the UFSAR. Under this approach, the referenced document is maintained on-site in accordance with licensee administrative processes, and changes are evaluated using 10 CFR 50.59.

Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," dated September 1999 (ADAMS Accession No. ML992930009), states that NEI 98-03, Revision 1, provides methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.71(e).

2.2.3 NRC Generic Letter 91-08

Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications" (ADAMS Accession No. ML031140516), provides guidance for removing component lists from the TSs and relocating them to other licensee-controlled documents. In GL 91-08, the NRC stated that a request to remove component lists from the TS should address the following:

1. Each TS should include an appropriate description of the scope of the components to which the TS requirements apply. Components that are defined by regulatory requirements or guidance need not be clarified further. However, the Bases Section of the TS should reference the applicable requirements or guidance.

2. If the removal of a component list results in the loss of notes that modify or provide an exception to the TS requirements, the specification should be revised to incorporate that modification or exception. The modification or exception should be stated in terms that identify any group of components by function rather than by plant identification number, if practical.

3. Licensees should confirm that the lists of components removed from the TS are located in appropriately controlled plant procedures. The list of components may be included in the next update of the FSAR. The Bases Section of individual specifications also may reference the plant procedures or other documents that identify each component list. The Bases Section for the containment isolation valve TS should be updated to describe the intent of opening valves under administrative control.

At the time of issuance, May 6, 1991, GL 91-08 specifically addressed the removal of PIV tables, stating:

> Guidance on removing from the TS the list of reactor coolant system pressure isolation valves is pending the NRC staff's resolution of generic concerns with existing lists for these valves. In the interim, licensees should not submit proposals to remove this list from the TS.

The NRC staff's generic concerns referenced above are related to interfacing system LOCAs and were designated generic safety issue 105. NUREG-1463, "Regulatory Analysis for the Resolution of Generic Safety Issue 105: Interfacing System Loss-of-Coolant Accident in Light-Water Reactors" (ADAMS Accession No. ML072500088), discusses the NRC staff's resolution of interfacing system LOCAs. As discussed in Sections 2.2.4 and 3.0 below, this issue has been resolved. Accordingly, GL 91-08 can now be used for removal of PIV lists from the TSs.

2.2.4 NUREG-1463

NUREG-1463 provides the NRC staff's regulatory analysis of generic safety issue 105, specifically, interfacing system LOCAs resulting from the failure or improper operation of two or more PIVs that compose the pressure boundary between the RCS and low pressure systems. It is concluded in NUREG-1463 that participation in the ongoing individual plant examination process was sufficient to resolve generic issue 105. This resolution was promulgated to licensees by NRC Information Notice 92-36, Supplement 1, "Intersystem LOCA Outside Containment" (ADAMS Accession No. ML031190681).

### 2.2.5 NUREG-1431

NUREG-1431, Revision 4, "Standard Technical Specifications – Westinghouse Plants" (ADAMS Accession No. ML12100A222), was used by the NRC staff as guidance regarding the PIV leakage requirements that should be included in the TSs. As discussed in the Bases for the Standard Technical Specifications, RCS PIVs are defined as "any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system." The TSs allow RCS high pressure operation when leakage through the valves exists in amounts that do not compromise safety. The RCS PIV leakage Standard Technical Specification does not include a table listing the individual valves. Therefore, removing the RCS PIV lists from the Salem, Unit Nos. 1 and 2, TSs, is in alignment with NUREG-1431.

### 2.3 Proposed TS Changes

The licensee has proposed to remove the RCS PIV lists from the Salem, Unit Nos. 1 and 2, TSs. The specific changes to each unit's TSs are described below.

### 2.3.1 Salem, Unit No. 1

- Relocate the list of valves from TS Table 4.4-3, "Reactor Coolant System Pressure Isolation Valves," to the Salem TRM. Relocate the PIV leakage limits and Notes (a) and (b) from the table to SR 4.4.6.3.

- Delete the reference to TS Table 4.4-3 from TS LCO 3.4.6.3, TS Action 3.4.6.3.a and SR 4.4.6.3.

### 2.3.2 Salem, Unit No. 2

- Relocate TS Table 3.4-1, "Reactor Coolant System Pressure Isolation Valve Function," in its entirety, to the Salem TRM.

- Delete the reference to TS Table 3.4-1 from TS LCO 3.4.7.2.f and SR 4.4.7.2.2.

Proposed changes to the Salem, Unit Nos. 1 and 2, TS Bases were provided for information only. The TS Bases are controlled by TS Bases Control Program (TS 6.17 (Unit No. 1) and TS 6.16 (Unit No. 2)), and any changes to the TS Bases will be made in accordance with that program.

## 3.0 TECHNICAL EVALUATION

The NRC staff issued GL 91-08 to give guidance on relocating component lists from the TSs to licensee-controlled documents. A license amendment to relocate these component lists allows licensees to revise or update the lists in the future without having to apply for another license amendment. Changes to such lists, when relocated to licensee-controlled documents, are subject to review under the provisions of 10 CFR 50.59; therefore, a formal means of control over these lists still exists.

Enclosure 1 to GL 91-08 contains specific items that should be addressed by any request to remove component lists from plant TSs. The three items to be addressed are discussed in Section 2.2.3 above.

With regard to item (1) above, PIVs are described in NUREG-1431 "as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system." The removal of the PIV tables from the Salem TSs does not change the scope of components to which the TSs apply. In addition, PSEG will update the Salem TSs Bases with a description of the PIVs.

With regard to item (2) above, the Salem, Unit No. 1, leakage acceptance criteria and associated notes are being relocated from Table 4.4-3 to SR 4.4.6.3. The Salem, Unit No. 2, PIV table does not contain any notes; therefore, this consideration does not apply.

With regard to item (3) above, the licensee stated in its September 27, 2017, letter, that the list of PIVs will be relocated to the TRM. The TRM qualifies as an appropriately controlled plant procedure.

As noted above, GL 91-08 states that licensees should not request removal of RCS PIV lists until the generic concerns regarding interfacing systems LOCAs are resolved. This generic concern has been resolved as addressed in NUREG-1463. Therefore, the licensee's request to remove the PIV tables from the Salem TSs is appropriate.

After reviewing the licensee's application, the NRC staff has determined that the proposed amendments do not change any requirements with respect to PIVs. The definition of what constitutes a PIV is already contained in NUREG-1431. The TS LCOs and SRs continue to apply to the PIVs. The PIV lists will be moved to the TRM, and future changes will be controlled by 10 CFR 50.59. The NRC staff finds that the requested changes will be made in accordance with the guidance in GL 91-08 and are consistent with NURG-1431. Therefore, the NRC staff finds the proposed changes acceptable.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments on January 9, 2018. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (82 FR 55408; November 21, 2017). 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 Contributor: Carleen Parker

Date: April 18, 2018

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 323 AND 304 RE: RELOCATION OF REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE TABLES (EPID L-2017-LLA-0313) DATED APRIL 18, 2018

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