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

September 4, 2015

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

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NO. 1 – ISSUANCE OF EMERGENCY AMENDMENT REGARDING REMOVAL OF PRESSURIZER POWER OPERATED RELIEF VALVE POSITION INDICATION INSTRUMENTATION FROM TECHNICAL SPECIFICATIONS (TAC NO. MF6669)

Dear Mr. Braun:

The Commission has issued the enclosed Amendment No. 310 to Renewed Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the technical specifications (TSs) in response to your application dated August 31, 2015, as supplemented by letter dated September 2, 2015.

The amendment removes the pressurizer power operated relief valve position indication from the accident monitoring instrumentation TSs and associated surveillance requirements.

A copy of the related safety evaluation (SE) is also enclosed. The SE describes the emergency circumstances under which the amendment was issued and the final determination of no significant hazards. Notice of Issuance, addressing the final no significant hazards determination and opportunity for a hearing, will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Carleen J. Parker, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-272

Enclosures:
1. Amendment No. 310 to Renewed DPR-70
2. Safety Evaluation

cc w/enclosures: Distribution via 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. 310
Renewed License No. DPR-70

1. The 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 August 31, 2015, as supplemented by letter dated September 2, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (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. 310, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: September 4, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 310

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 revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

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

Remove	Insert
3/4 3-54	3/4 3-54
3/4 3-55	3/4 3-55
3/4 3-57a	3/4 3-57a

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. 310, 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

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1	1, 2
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1	1, 2
3. Reactor Coolant Pressure (Wide Range)	2	1	1, 2
4. Pressurizer Water Level	2	1	1, 2
5. Steam Line Pressure	2/Steam Generator	1/Steam Generator	1, 2
6. Steam Generator Water Level (Narrow Range)	2/Steam Generator	1/Steam Generator	1, 2
7. Steam Generator Water Level (Wide Range)	4 (1/Steam Generator)	3 (1/Steam Generator)	1, 2
8. Refueling Water Storage Tank Water Level	2	1	1, 2
9. deleted			
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	3 (1/Steam Generator)	4, 6
11. Reactor Coolant System Subcooling Margin Monitor	2	1	1, 2
12. Deleted			

TABLE 3.3-11 (CONTINUED)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
13. PORV Block Valve Position Indicator	2/valve**	1	1, 2
14. Pressurizer Safety Valve Position Indicator	2/valve**	1	1, 2
15. Containment Pressure - Narrow Range	2	1	1, 2
16. Containment Pressure - Wide Range	2	1	7, 2
17. Containment Water Level - Wide Range	2	1	7, 2
18. Core Exit Thermocouples	4/core quadrant	2/core quadrant	1, 2
19. Reactor Vessel Level Instrumentation System (RVLIS)	2	1	8, 9
20. Containment High Range Accident Radiation Monitor	2	2	10
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor	1/MS Line	1/MS Line	10

(**) Total number of channels is considered to be two (2) with one (1) of the channels being any one (1) of the following alternate means of determining PORV Block, or Safety Valve position: Tailpipe Temperatures for the valves, Pressurizer Relief Tank Temperature Pressurizer Relief Tank Level OPERABLE.

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK[(1)]	CHANNEL CALIBRATION[(1)]	CHANNEL FUNCTIONAL TEST[(1)]
12. Deleted			
13. PORV Block Valve Position Indicator		N.A.	*
14. Pressurizer Safety Valve Position Indicator		N.A.	
15. Containment Pressure - Narrow Range			N.A.
16. Containment Pressure - Wide Range			N.A.
17. Containment Water Level - Wide Range			N.A.
18. Core Exit Thermocouples			N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)			N.A.
20. Containment High Range Accident Radiation Monitor			
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor			

Table Notation

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

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* Unless the block valve is closed in order to meet the requirements of Action b, or c in specification 3.4.3.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 310

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-70

PSEG NUCLEAR, LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-272

1.0 INTRODUCTION

By letter dated August 31, 2015, as supplemented by letter dated September 2, 2015 (Agencywide Document Access and Management System (ADAMS) Accession Nos. ML15243A491 and ML15245A754), PSEG Nuclear, LLC (PSEG or the licensee) submitted a request for an emergency license amendment to the Salem Nuclear Generating Station (Salem), Unit No. 1, Technical Specifications (TSs). The amendment removes the pressurizer power operated relief valve (PORV) position indication instrumentation from the accident monitoring instrumentation TSs and associated surveillance requirements (SRs).

2.0 REGULATORY EVALUATION

The Commission's regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TSs include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in plant TSs.

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

As discussed in 10 CFR 50.36(c)(3), SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, facility operation will be within safety limits, and the LCOs will be met.

On July 22, 1993 (58 *Federal Register* (FR) 39132), the Commission published a "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 codified in 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:

> <u>Criterion 1</u>
> Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> <u>Criterion 2</u>
> 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.
>
> <u>Criterion 3</u>
> 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.
>
> <u>Criterion 4</u>
> 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.

NUREG-1431, Revision 4, "Standard Technical Specifications Westinghouse Plants," dated April 2012 (ADAMS Accession No. ML12100A228), was developed based on the criteria in the Final Policy Statement.

Salem, Unit No. 1, was designed using the Atomic Industry Forum General Design Criteria (GDC) and the licensee's understanding of the intent of the Atomic Energy Commission's (AECs) proposed GDC. A comparison of the Salem, Unit Nos. 1 and 2, plant design was done with 10 CFR Part 50, Appendix A, GDC for Nuclear Power Plants, dated July 7, 1971. The comparison was documented in the Salem Updated Final Safety Analysis Report (UFSAR), Section 3.1.3, which states, in part, that, "The Salem Plant design conforms with the intent of the 'General Design Criteria for Nuclear Power Plants,' dated July 7, 1971...."

AEC proposed GDC 16, "Monitoring Reactor Coolant Pressure Boundary," requires monitoring of the reactor coolant pressure boundary (RCPB) to detect leakage. Criterion 16 of the

proposed AEC GDC is the equivalent of the 10 CFR, Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary."

The U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," issued December 1980 (ADAMS Accession No. ML060750525), lists five types (Types A–E) of variables to help designers select the accident monitoring instrumentation and applicable criteria. Categories 1, 2, and 3 separate the type of criteria into groups for a graded approach to requirements, depending on the importance to safety or the measurement of a specific variable.

As required by 10 CFR 50.34, "Contents of applications; technical information," each applicant for an operating license shall include a final safety analysis report. The UFSAR incorporates changes made to the FSAR in accordance with 10 CFR 50.71, "Maintenance of records, making reports," Section (e). Section 50.71(e) of 10 CFR also requires a licensee to submit revisions to the NRC on a periodic basis, not to exceed 24 months. Changes to the UFSAR are controlled by 10 CFR 50.59, "Changes, tests and experiments."

The NRC staff also considered the following guidance for its review:

- NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.
- NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980, Item II.D.3.

3.0 TECHNICAL EVALUATION

3.1 System Description

The reactor coolant system (RCS) is protected against over-pressurization by control and protective circuits such as the pressurizer pressure high reactor trip and by the PORVs (1 PR1 and 1 PR2) connected to the top of the pressurizer. The PORVs provide the means for pressure relief. Venting through the PORVs can limit the pressurizer pressure to levels below the pressurizer safety valve set pressure, and thereby avoid opening (or challenging) the pressurizer safety valves. Discharge from the PORVs is sent to the pressurizer relief tank (PRT).

Each PORV is equipped with two limit switches to provide an alarm and an indication (i.e. light) in the control room if the PORV is not fully closed. The PORV position instrumentation does not provide an input to any automatic trip function or impact the response of the PORVs to a design-basis accident. Furthermore, this instrumentation is not needed for manual operator action necessary for safety systems to accomplish their safety function for the design-basis events.

3.2 Proposed TS Change

Salem, Unit No. 1, TS 3/4.3.3.7, "Accident Monitoring Instrumentation," Tables 3.3-11 and 4.3-11, list the instrumentation necessary for accident monitoring. The associated LCO 3.3.7, requires that the accident monitoring instrumentation channels shown in Table 3.3-11 shall be

operable. Instrument 12 in Table 3.3-11 requires two channels per valve be operable for the PORV position indicator (PI). The required number of channels is modified by a footnote ** that states the "[t]otal number of channels is considered to be two (2) with one (1) of the channels being any one (1) of the following alternate means of determining PORV, PORV Block, or Safety Valve position: Tailpipe Temperatures of the valves, Pressurizer Relief Tank temperature[,] Pressurizer Relief Tank Level OPERABLE." The associated SR 4.3.3.7 states that each accident monitoring instrumentation channel shall be demonstrated operable by performance of the channel check, channel calibration, and channel functional test operations at the frequencies specified in the Surveillance Frequency Control Program, unless otherwise noted in Table 4.3-11.

The licensee is requesting to remove PORV PIs from the Salem, Unit No. 1, TSs, which correspond to item 12 in both Tables 3.3-11 and 4.3-11. Removal of this item will also require revision of the footnote ** on page 3/4 3-55 to delete reference to PORV indication.

3.3 Commitment to NUREG-0737

NUREG-0737, item II.D.3, requires that reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe. In Appendix A, item 2.1.3.a of the Salem UFSAR, the licensee indicates how it complies with this requirement. Specifically, the licensee notes each PORV includes limit switches to indicate the open and closed positions of the valves in the control room. NUREG-0737, item II.D.3, also points out that direct valve position indication (or acoustic monitors) should be part of the plant TSs. In response to the NUREG-0737 item, the PORV PIs were included in the Salem, Unit No. 1, TSs. The licensee states in its August 31, 2015, and September 2, 2015, letters, that the current regulation in 10 CFR 50.36 clarifies those items that are required to be contained in the plant TSs and that the PORV position indication can be removed from Salem, Unit No. 1, TSs.

In the licensee's request, the licensee noted that the PORV PI will only be removed from the TSs, but the instrument will continue to provide indication in the control room. No physical modification to the plant is requested. Furthermore, in the supplement dated September 2, 2015, the licensee confirmed that the PI will remain in the UFSAR, Section 7.5, "Safety-Related Display Instrumentation"; Table 7.5-4, "Summary of Instrumentation Compliance with RG 1.97"; and in UFSAR Appendix A, "TMI Lessons Learned," Section 2.1.3.a, "Relief and Safety Valve Position Indications."

The licensee also indicated that surveillance of the PORVs PIs are part of Salem, Unit No. 1, surveillance procedures. Furthermore, operability of the PORV PIs will be evaluated in accordance with current NRC guidance found in Inspection Manual Chapter 0326, "Operability Determinations & Functionality Assessments for Conditions Adverse to Quality or Safety." Specifically, when an SSC is determined to be nonfunctional, operations shift management is responsible to determine any compensatory measures or corrective actions required and any impact on supported TS functions. In addition, the licensee indicated functionality of the PORV position indication will continue to be demonstrated in conjunction with the PORV surveillance testing in accordance with TS 3.4.3, "Reactor Coolant System Relief Valves"; 3.4.9.3, "Reactor Coolant System Overpressure Protection Systems"; and 6.8.4.j, "Inservice Testing Program." Based on the information provided, the NRC staff finds the licensee will continue to perform

surveillance of the PORVs PI in accordance with plant procedures, even after removal of the indication from Salem, Unit No. 1, TSs.

Based on the above, the NRC staff determined that Salem, Unit No. 1 will continue to meet its commitment to the requirements in NUREG-0737, item II.D.3.

3.4 NRC Staff Evaluation

The licensee analyzed the safety basis for PORV PI using the Final Policy Statement's four criteria codified by 10 CFR 50.36. Specifically, the licensee states that, "[t]he PORVs themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a Design Basis Accident (DBA), and therefore, meet Criterion 3 of the NRC Final Policy Statement." Accordingly, their operability is required by TS 3.4.3, "Relief Valves." However, the licensee states that PORV position indication is not relied upon as the primary means to detect or indicate a significant abnormal degradation of the RCPB as outlined by Criterion 1.

In its letter dated August 31, 2015, the licensee states, "PORV position indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2."

The licensee indicated that while the function of the PORVs is part of the primary success path for accident prevention or mitigation in the UFSAR, the PORV position indication function is not part of the primary success path. UFSAR accident analyses (e.g. the Spurious Operation of The Safety Injection System at Power) assumes that the PORVs open as designed to reduce reactor pressure, and no operator action based on PORV position indication is required. Therefore, PORV position indication is not part of the primary success path as indicated in Criterion 3. Furthermore, the licensee stated that the loss of this instrumentation has no effect on the probabilistic safety assessment (PSA) and has not been shown to be significant to public health and safety as considered in Criterion 4.

The NRC staff reviewed each of the criteria in 10 CFR 50.36 for applicability to the PORV position indication instrumentation. The PORV position indication instrumentation is not used to detect and indicate in the control room a significant abnormal degradation of the RCPB.

Criterion 1 of 10 CFR 50.36(a)(2)(ii) specifies that LCOs be established for installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the RCPB. The Final Policy Statement states that this criterion is intended to ensure that TSs control those instruments specifically installed to detect excessive RCS leakage. This criterion should not be interpreted to include instrumentation to detect precursors to RCPB leakage, instrumentation to identify the source of actual leakage, seismic instrumentation, loose parts monitor, or valve PIs. The primary equipment used to detect a significant abnormal degradation of the RCPB is provided in Salem, Unit No. 1, TSs 3.3.3.1, "Radiation Monitoring Instrumentation," and 3.4.6.1, "Reactor Coolant System Leakage – Leakage Detection Systems." The specific equipment and functions are the containment atmosphere particulate radioactivity monitoring system, the containment sump monitoring instrumentation, the containment fan cooler condensate flow, and the containment atmosphere gaseous radioactivity monitoring system. The PORV PIs are not included in these TSs. In

addition, the staff reviewed the Salem UFSAR and determined that the PORV PIs are not credited for indicating a significant abnormal degradation of the RCPB in the accident analysis. Therefore, it is concluded that the PORV position indication instrumentation does not meet Criterion 1 for inclusion in the Salem, Unit No. 1, TSs.

Criterion 2 of 10 CFR 50.36(c)(2)(ii) specifies that LCOs be established for process variables, design features, or operating restrictions that are an initial condition of a design-basis accident (DBA) or transient analysis that either assumes the failure of, or presents a challenge to, fission product barrier integrity. The Final Policy Statement on TSs explains that the purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. It includes active design features and operating restrictions needed to preclude unanalyzed accidents and transients.

The PORV position indication is not described in the UFSAR as providing indication of an initial condition of a DBA or transient analysis. The function of the PORV itself is included in the analysis of transients and accidents in Chapter 15 of the UFSAR. However, the licensee stated in its letter dated September 2, 2015, in response to the NRC staff's request for additional information (RAI), that the PORV position indication is not the only indication used by the operators to determine PORV position. The alternate means include valve tailpipe temperatures, PRT temperature, and PRT level. Based on the above, the NRC staff determined that the PORV position indication does not meet Criterion 2 for inclusion in the Salem, Unit No. 1, TSs.

Criterion 3 of 10 CFR 50.36(c)(2)(ii) specifies that LCOs be established for a structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. The licensee stated in its letter dated August 31, 2015, that the function of the PORV is part of the primary success path in the UFSAR; however, the PORV position indication is not part of the primary success path. The licensee also stated that the accident analysis assumes that the PORVs open as designed to reduce reactor pressure, and no operator action based on PORV position indication is required. Further, the licensee stated that the individual Emergency Operating Procedures (EOPs) using PORV position indication can be accomplished using alternate means, regardless of whether PORV position instrumentation is available. In its August 31, 2015, and September 2, 2015, letters, the licensee noted that the EOPs include steps to verify whether the PORVs are closed and determine if they are operating properly (e.g., 1-EOP-TRIP-1). If the PORVs should be closed but they are not, instructions are provided that they be closed. If the operator action to close them is unsuccessful, the associated block valve(s) are closed. Indications of these variables are provided in the control room. In response to the NRC staff's RAI, the licensee stated in its letter dated September 2, 2015, that the operator will use the limit switch indication, but will also verify PORV position by alternate means (valve tailpipe temperature, PRT temperature, and PRT level), as needed, to determine PORV position. These indications are surveilled monthly

Accident monitoring instrumentation is provided to monitor variables and systems over their anticipated ranges for accident conditions, as appropriate, to ensure adequate safety during and following accidents. These variables are used by the control room operating personnel to perform their role in the emergency plan in the evaluation, assessment, and monitoring of events

and execution of control room functions. RG 1.97 provides criteria for determining acceptable quality standards for post-accident monitoring instrumentation based on the categorization of the instruments.

The licensee stated in its letter dated August 31, 2015, that PORV position indication at Salem is classified as an RG 1.97, Type D, Category 2, instrument. RG 1.97 describes Category 2, Type D, variables as follows:

> Category 2 provides for qualification, but it is less stringent because it does not include seismic qualification, redundancy, or continuous display and because it requires only a high-reliability power source (not necessarily standby power).
>
> Type D instruments provide information to indicate the operation of individual safety systems and other systems important to safety.

RG 1.97, Type A, instruments monitor primary information required to permit the control room operator to take specific manually-controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBA events. RG 1.97, Category 1, instruments are designed for full qualification, redundancy, continuous real-time display, and onsite (standby) power. NUREG-1431 includes operability requirements for post-accident monitoring instrumentation that meet the definition of Type A and Category 1, non-Type A instruments, because these instruments provide key information required by control room operators to perform manual actions to mitigate events. As such, NUREG-1431 does not include operability requirements for a PORV PI classified as a RG 1.97, Type D, Category 2, instrument.

Based on the NRC staff's review of the Salem UFSAR, RG 1.97, and NUREG-1431, the staff concluded that PORV position indication is not part of the primary success path and which functions or actuates to mitigate a DBA or transient; therefore, the indication does not meet the requirements of Criterion 3. This conclusion is consistent with the guidance for the content of Standard Technical Specifications (STSs).

Criterion 4 of 10 CFR 50.36(c)(2)(ii) requires that an LCO be established for an SSC, which operating experience or PSA has shown to be significant to public health and safety. Based on the discussion above, the NRC staff has determined that the loss of PORV position indication instrumentation has not been shown to be significant to public health and safety. Based on this, the NRC staff concludes that the PORV position indication instrumentation does not meet Criterion 4. This conclusion is consistent with the guidance for the content of STSs.

Based on the reasons described above, the NRC staff has determined that PORV position indication instrumentation does not meet any of the four screening criteria of the Final Policy Statement for Salem, Unit No. 1. As such, the deletion of the operability requirements for the PORV position indication, as well as modification of the footnote ** to delete the reference to PORV position and deletion of the line item in the associated SR Table 4.3-11 are acceptable.

3.4 NRC Staff Conclusion

The NRC staff reviewed the licensee's request related to the removal of the PORV PIs from Salem, Unit No. 1, TS 3/4.3.3.7, "Accident Monitoring Instrumentation," and Tables 3.3-11 and 4.3-11, including the associated footnotes, against the analyses in the Salem UFSAR and the applicability of each of the criteria provided in 10 CFR 50.36(c)(2)(ii) to determine if the PORV position indication meets the requirements for inclusion in the TSs. Based on the above evaluation, the NRC staff concludes that the PORV position indication instrumentation is not required to satisfy any of the four criteria in 10 CFR 50.36(c)(2)(ii) for items required to be maintained in the TSs, the proposed changes are consistent with the guidance in NUREG-1431 and the guidelines in RG 1.97, and that the licensee continues to meet its commitment to the requirements in NUREG-0737, item II.D.3 for the PORV position indication. In addition, the pressurizer PORV position indication will remain in the Salem UFSAR. Therefore, the licensee will evaluate any future changes against the criteria in 10 CFR 50.59 or will submit the changes to the NRC for prior approval.

Therefore, the NRC staff concludes removal of the PORV position indication instrumentation from Salem, Unit No. 1 TSs is acceptable.

4.0 EMERGENCY CIRCUMSTANCES

The NRC's regulations in 10 CFR 50.91(a)(5) state that where the NRC finds that an emergency situation exists in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the NRC may issue a license amendment involving no significant hazards consideration (NSHC) without prior notice and opportunity for a hearing or for public comment. In such a situation, the NRC will publish a notice of issuance under 10 CFR 2.106, providing for opportunity for a hearing and for public comment after issuance.

As discussed in the licensee's application dated August 31, 2015, as supplemented by letter dated September 2, 2015, the licensee requested that the proposed amendment be processed by the NRC on an emergency basis. The licensee stated that the emergency circumstance could not be avoided, "because the discovery of the inoperable PORV position indication could not have been foreseen in sufficient time to allow the 30 day public comment period specified in 10 CFR 50.91(2)(ii)." By letter dated July 31, 2015 (ADAMS Accession No. ML15217A309), Dynapar Corporation notified PSEG that some Namco EA180 and EA170 limit switches manufactured between March 25, 2014, and December 30, 2014, may contain inadequately stress-relieved return springs, preventing the normally closed contacts from returning to their initial position. This could result in an incorrect signal on the position of the valve being monitored. PSEG evaluated the vendor information, and concluded that, "the potential for failure of the position switches on the 1PR1 and 1PR2 are such that a reasonable expectation of OPERABILITY cannot be established." Based on PSEG's conclusion, the action statement for TS 3.3.3.7, "Accident Monitoring Instrumentation," was entered.

Salem, Unit No. 1, was operating in MODE 1 at the time PSEG received the vendor report and at the time PSEG concluded that a reasonable expectation of OPERABILITY could not be established. PORV PIs 1PR1 and 1PR2 are contained in TS 3.3.37, Table 3.3-11, "Accident Monitoring Instrumentation." TS Table 3.3-11, Action 1, requires the inoperable accident

monitoring channel to be restored to OPERABLE status within 7 days or for Salem, Unit No. 1, to be in HOT SHUTDOWN within the next 12 hours. Salem, Unit No. 1, entered the action statement on August 28, 2015, at 1630 hours; therefore, PSEG is required to restore the position switch to OPERABLE status by September 4, 2015, at 1630 or be in HOT SHUTDOWN within the following 12 hours.

With regards to the option to conduct repair activities while online, PSEG stated:

> The PORVs are located in the containment in an area in which high temperatures significantly limit the stay-time for maintenance personnel impacting personnel safety. In addition, replacement of the affected position switch would require PORVs to be stroked for retesting, potentially subjecting the plant to an unnecessary operational transient."

The NRC staff reviewed the licensee's basis for processing the proposed amendment as an emergency amendment and finds that an emergency situation exists consistent with the provisions in 10 CFR 50.91(a)(5). Furthermore, the NRC staff determines that (1) the licensee used its best efforts to make a timely application; (2) the licensee could not reasonably have avoided the situation; and (3) the licensee has not abused the provisions of 10 CFR 50.91(a)(5). Based on these findings and the determination that the amendment involves NSHC as discussed below, the NRC staff has determined that a valid need exists for issuance of the license amendment using the emergency provisions of 10 CFR 50.91(a)(5).

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC's regulation in 10 CFR 50.91(a)(5) states that where the Commission finds that an emergency situation exists, the Commission will not publish a notice of propose determination on NSHC. The NRC's regulations in 10 CFR 50.92 state that the NRC may make a final determination that a license amendment involves NSHC if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As discussed in Section 4.0 of this SE, the NRC staff finds that an emergency situation exists for Salem, Unit No. 1. Therefore, as provided below, the NRC has made a final determination concerning whether the license amendment involves a significant hazards consideration.

As required by 10 CFR 50.91(a), PSEG has provided its analysis of the issue of NSHC, which is presented below. The underlining represents information that was added to the original analysis by PSEG in its supplement dated September 2, 2015.

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change to the TS would remove the PORV position indication from the Accident Monitoring Instrumentation TS for Salem

Unit 1. The failure of this instrumentation is not assumed to be an initiator of any analyzed event in the UFSAR. Therefore the probability of an accident previously evaluated is not significantly increased.

The proposed changes do not alter the design of the PORVs or any other system, structure, or component (SSC). The proposed changes conform to NRC regulatory guidance regarding the content of plant TS, as identified in 10 CFR 50.36, NUREG-1431, and the NRC Final Policy Statement in 58 FR 39132. TS Operability requirements are retained for Type A and Category 1 variables. Operability of these instruments ensures sufficient information is available to monitor and assess plant status during and following an accident. Alternate means for diagnosing and responding to PORV malfunctions (Pressurizer Relief Tank level and temperature, and PORV tailpipe temperature) are unaffected by the proposed change. Therefore, the consequences of an accident previously evaluated are not significantly increased.

Therefore, these proposed changes do not represent a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes to the TS would remove the PORV position indication from the Accident Monitoring Instrumentation TS for Salem Unit 1. The proposed change does not involve a modification to the physical configuration of the plant or change in the methods governing normal plant operation. The proposed changes will not impose any new or different requirement or introduce a new accident initiator, accident precursor, or malfunction mechanism.

   Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes to the TS would remove the PORV position indication from the Accident Monitoring Instrumentation TS for Salem Unit 1. This instrumentation is not needed for manual operator action necessary for safety systems to accomplish their safety function for the design basis events. The

> PORV position instrumentation does not provide an input to any automatic trip function or impact the response of the PORVs to a design basis accident.
>
> Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed PSEG's analysis and, based on this review, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that NSHC is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 because the PORV position indicator instrumentation only provides information about the position of the valve, its removal involves 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 made a finding in this document that the amendment involves NSHC. Accordingly, the amendment meets 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 amendment.

8.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: R. Alvarado
M. Chernoff
C. Parker

Date: September 4, 2015

September 4, 2015

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

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NO. 1 – ISSUANCE OF EMERGENCY AMENDMENT REGARDING REMOVAL OF PRESSURIZER POWER OPERATED RELIEF VALVE POSITION INDICATION INSTRUMENTATION FROM TECHNICAL SPECIFICATIONS (TAC NO. MF6669)

Dear Mr. Braun:

The Commission has issued the enclosed Amendment No. 310 to Renewed Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the technical specifications (TSs) in response to your application dated August 31, 2015, as supplemented by letter dated September 2, 2015.

The amendment removes the pressurizer power operated relief valve position indication from the accident monitoring instrumentation TSs and associated surveillance requirements.

A copy of the related safety evaluation (SE) is also enclosed. The SE describes the emergency circumstances under which the amendment was issued and the final determination of no significant hazards. Notice of Issuance, addressing the final no significant hazards determination and opportunity for a hearing, will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Carleen J. Parker, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-272

Enclosures:
1. Amendment No. 310 to Renewed DPR-70
2. Safety Evaluation

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ADAMS Accession No.: ML15245A636 *by email dated 9/4/15

OFFICE	NRR/DORL/LPL1-2/PM	NRR/DORL/LPL1-2/LA	NRR/DSS/STSB/BC*	NRR/DSS/SRXB/BC
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DATE	9/3/15	9/4/15	9/3/15*	9/3/15
OFFICE	NRR/DE/EICB/BC*	OGC	NRR/DORL/LPL1-2/BC	NRR/DORL/LPL1-2/PM
NAME	DRahn	BMizuno w/comments	DBroaddus	CParker
DATE	9/3/15	9/4/15	9/4/15	9/4/15

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