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

June 28, 2017

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 AMENDMENTS TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-545, REVISION 3, "TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5 TESTING" (CAC NOS. MF8311 AND MF8312)

Dear Mr. Sena:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment Nos. 319 and 300 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 August 30, 2016.[1]

The amendments approve adoption of NRC-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015.[2]

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,

Richard B. Ennis, 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. 319 to Renewed DPR-70
2. Amendment No. 300 to Renewed DPR-75
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML16243A233
[2] ADAMS Accession No. ML15294A555

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-545, REVISION 3, "TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5 TESTING" (CAC NOS. MF8311 AND MF8312) DATED JUNE 28, 2017

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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. 319
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 August 30, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 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. 319, 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 G. 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: June 28, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 319

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.

Remove	Insert
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
I	I
1-4	1-4
3/4 1-10	3/4 1-10
3/4 1-11	3/4 1-11
3/4 4-4	3/4 4-4
3/4 4-4a	3/4 4-4a
3/4 4-5a	3/4 4-5a
3/4 4-16a	3/4 4-16a
3/4 4-31	3/4 4-31
3/4 5-5a	3/4 5-5a
3/4 6-9	3/4 6-9
3/4 6-13	3/4 6-13
3/4 7-1	3/4 7-1
3/4 7-10	3/4 7-10
3/4 9-8a	3/4 9-8a
6-19e	6-19e

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. 319, 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

SECTION PAGE

1.0 DEFINITIONS

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system (primary-to-secondary leakage).

INSERVICE TESTING PROGRAM

1.15.1 The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall be all those persons who are not occupationally associated with the plant. This category does not include employees of PSE&G, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent controls and Radiological Environmental Monitoring programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8 respectively.

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.#

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

\# A maximum of one centrifugal charging pump shall be OPERABLE while in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to 312°F, MODE 5, or MODE 6 when the head is on the reactor vessel.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

3/4.4.2 SAFETY VALVES

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE* with a lift setting of 2485 psig ± 3%.**,***

APPLICABILITY: MODE 4 and 5

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

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* While in Mode 5, an equivalent size vent pathway may be used provided that the vent pathway is not isolated or sealed.

** The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

*** Following testing the lift setting shall be reset to within ± 1%.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 3%.*,**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes, or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

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* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Following testing the lift setting shall be reset to within ± 1%.

3/4.4.3 RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.3.1 In addition to the requirements of the INSERVICE TESTING PROGRAM, each PORV shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

b. Operating solenoid valves, air control valves, and check valves on associated air accumulators in PORV control systems through one complete cycle of full travel, and

c. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.3.2 Each block valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b, or c in Specification 3.4.3.

REACTOR COOLANT SYSTEM

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor Coolant System Pressure Isolation Valves specified in table 4.4-3 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 in Table 4.4-3, 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 specified in Table 4.4-3 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 listed in Table 4.4-3 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

4.4.9.3.1 Each POPS shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the POPS actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the POPS is required OPERABLE, and in accordance with the Surveillance Frequency Control Program thereafter when the POPS is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel in accordance with the Surveillance Frequency Control Program.

c. Verifying the POPS isolation valve is open in accordance with the Surveillance Frequency Control Program when the POPS is being used for overpressure protection.

d. Testing pursuant to the INSERVICE TESTING PROGRAM.

4.4.9.3.2 The RCS vent(s) shall be verified to be open in accordance with the Surveillance Frequency Control Program* when the vents(s) is being used for overpressure protection.

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* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open in accordance with the Surveillance Frequency Control Program.

f. By verifying that each of the following pumps develops the indicated Total Dynamic Head (TDH) when tested at the test flow point pursuant to the INSERVICE TESTING PROGRAM:

1. Centrifugal charging pump ≥ 2338 psi TDH

2. Safety Injection Pump ≥ 1369 psi TDH

3. Residual heat removal pump ≥ 165 psi TDH

g. By verifying the correct position of each of the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. In accordance with the Surveillance Frequency Control Program.

HPSI SYSTEM VALVE NUMBER	LPSI SYSTEM VALVE NUMBER
11 SJ 16	11 SJ 138
12 SJ 16	12 SJ 138
13 SJ 16	13 SJ 138
14 SJ 16	14 SJ 138
	11 SJ 143
	12 SJ 143
	13 SJ 143
	14 SJ 143

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1. For Safety Injection pumps, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is ≥ 453 gpm; and

b) The total flow rate through all four injection lines is ≤ 647 gpm, and

c) The difference between any pair of injection line flow rates is ≤ 12.0 gpm, and

d) The total pump flow rate is ≤ 664 gpm in the cold leg alignment, and

e) The total pump flow rate is ≤ 654 gpm in the hot leg alignment.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and transferring suction to the RHR pump discharge.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 204 psid when tested pursuant to the INSERVICE TESTING PROGRAM.

c. In accordance with the Surveillance Frequency Control Program during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment High-High pressure test signal.

   2. Verifying that each spray pump starts automatically on a Containment High-High pressure test signal.

d. Following activities that could result in nozzle blockage, either evaluate the work performed to determine the impact to the containment spray system, or perform an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Not used.

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each required Purge and each Pressure-Vacuum Relief valve actuates to its isolation position.

e. Verifying that the Containment Pressure-Vacuum Relief Isolation valves are limited to $\leq$ 60% opening angle.

4.6.3.1.3 In accordance with the Surveillance Frequency Control Program, verify that on a main steam isolation test signal, each main steam isolation valve actuates to its isolation position.

4.6.3.1.4 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

4.6.3.1.5 Each required containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then in accordance with the Surveillance Frequency Control Program, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2.b for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60La.

4.6.3.1.6 A pressure drop test to identify excessive degradation of resilient valve seals shall be conducted on the:

a. Required Containment Purge Supply and Exhaust Isolation Valves in accordance with the Surveillance Frequency Control Program.

b. Deleted.

4.6.3.1.7 The required containment purge supply and exhaust isolation valves shall be determined closed in accordance with the Surveillance Frequency Control Program.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves (MSSVs) associated with each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or two main steam line code safety valves inoperable in one or more steam generators, operation in Modes 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or reduce power to less than or equal to the applicable percent of RATED THERMAL POWER per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With three main steam line code safety valves inoperable in one or more steam generators, operation in Modes 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valves are restored to OPERABLE status or reduce power to less than or equal to the applicable percent of RATED THERMAL POWER per Table 3.7-1 and within 36 hours, reduce the Power Range Neutron Flux High trip setpoint to less than or equal to the RATED THERMAL POWER per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Verify each required MSSV lift setpoint per Table 4.7-1. No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

otherwise, be in MODE 2 within the next 6 hours.

MODES 2 - and 3 With one or more main steam line isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may proceed provided;

a. The isolation valve(s) is (are) maintained closed, and

b. The isolation valve(s) is (are) verified closed once per 7 days.

Otherwise, be in MODE 3, HOT STANDBY, within the next 6 hours, and MODE 4, HOT SHUTDOWN, within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops operable, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per the INSERVICE TESTING PROGRAM.

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* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

3. If crack indications are found in portions of the SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage.

6.8.4.j Deleted

6.8.4.k Reactor Coolant Pump Flywheel Inspection Program

In addition to the requirements of the ISI Program, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

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. 300
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 August 30, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 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-75 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. 300, 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 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: June 28, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 300

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.

Remove	Insert
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
I	I
1-4	1-4
3/4 1-9	3/4 1-9
3/4 1-10	3/4 1-10
3/4 4-5	3/4 4-5
3/4 4-6	3/4 4-6
3/4 4-8a	3/4 4-8a
3/4 4-18	3/4 4-18
3/4 4-32	3/4 4-32
3/4 5-6	3/4 5-6
3/4 6-10	3/4 6-10
3/4 6-15	3/4 6-15
3/4 7-1	3/4 7-1
3/4 7-10	3/4 7-10
3/4 9-9	3/4 9-9
6-19f	6-19f

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

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

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 300, 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.

SECTION PAGE

1.0 DEFINITIONS

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system (primary-to-secondary leakage).

INSERVICE TESTING PROGRAM

1.15.1 The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall be all those persons who are not occupationally associated with the plant. This category does not include employees of PSE&G, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent controls and Radiological Environmental Monitoring programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8 respectively.

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s), and when all necessary attendant instrumentation, controls, normal or emergency electrical power source, cooling and seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.#

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

A maximum of one centrifugal charging pump shall be OPERABLE while in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to 312°F, MODE 5, or MODE 6 when the head is on the reactor vessel.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE* with a lift setting of 2485 psig ± 3%.**,***

APPLICABILITY: Mode 4 and 5

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

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* While in Mode 5, an equivalent size vent pathway may be used provided that the vent pathway is not isolated or sealed.

** The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

*** Following testing the lift setting shall be reset to within ± 1%.

REACTOR COOLANT SYSTEM

3/4.4.3 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 3%.*,**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

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* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Following testing the lift setting shall be reset to ± 1%.

SURVEILLANCE REQUIREMENTS

4.4.5.1 In addition to the requirements of the INSERVICE TESTING PROGRAM, each PORV shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

b. Operating solenoid valves, air control valves, and check valves on associated air accumulators in PORV control systems through one complete cycle of full travel, and

c. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.5.2 Each block valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b, or c in Specification 3.4.5.

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 specified in Table 3.4-1 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 listed in Table 3.4-1 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.

SURVEILLANCE REQUIREMENTS (Continued)

a. Performance of a CHANNEL FUNCTIONAL TEST on the POPS actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the POPS is required OPERABLE and in accordance with the Surveillance Frequency Control Program thereafter when the POPS is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel in accordance with the Surveillance Frequency Control Program.

c. Verifying the POPS isolation valve is open in accordance with the Surveillance Frequency Control Program when the POPS is being used for overpressure protection.

d. Testing pursuant to the INSERVICE TESTING PROGRAM.

4.4.10.3.2 The RCS vent(s) shall be verified to be open in accordance with the Surveillance Frequency Control Program* when the vent(s) is being used for overpressure protection.

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* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open in accordance with the Surveillance Frequency Control Program.

f. By verifying that each of the following pumps develops the indicated Total Dynamic Head (TDH) when tested at the test flow point pursuant to the INSERVICE TESTING PROGRAM:

1.	Centrifugal Charging pump	≥ 2338 psi TDH
2.	Safety Injection pump	≥ 1369 psi TDH
3.	Residual Heat Removal pump	≥ 165 psi TDH

g. By verifying the correct position of each of the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. In accordance with the Surveillance Frequency Control Program.

HPSI System Valve Number	LPSI System Valve Number
21 SJ 16	21 SJ 138
22 SJ 16	22 SJ 138
23 SJ 16	23 SJ 138
24 SJ 16	24 SJ 138
	21 SJ 143
	22 SJ 143
	23 SJ 143
	24 SJ 143

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1. For Safety Injection pumps, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is ≥ 453 gpm, and

b) The total flow rate through all four injection lines is ≤ 647 gpm, and

c) The difference between any pair of injection line flow rates is ≤ 12.0 gpm, and

d) The total pump flow rate is ≤ 664 gpm in the cold leg alignment, and

e) The total pump flow rate is ≤ 654 gpm in the hot leg alignment.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and transferring suction to the RHR pump discharge.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 204 psid when tested pursuant to the INSERVICE TESTING PROGRAM.

c. In accordance with the Surveillance Frequency Control Program during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment High-High pressure test signal.

   2. Verifying each spray pump starts automatically on a Containment High-High pressure test signal.

d. Following activities that could result in nozzle blockage, either evaluate the work performed to determine the impact to the containment spray system, or perform an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. NOT USED

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each required Purge and each Pressure-Vacuum Relief valve actuates to its isolation position.

e. Verifying that the Containment Pressure-Vacuum Relief Isolation valves are limited to $\leq 60°$ opening angle.

4.6.3.3 In accordance with the Surveillance Frequency Control Program, verify that on a main steam isolation test signal, each main steam isolation valve actuates to its isolation position.

4.6.3.4 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

4.6.3.5 Each required containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then in accordance with the Surveillance Frequency Control Program, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2.b for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60La.

4.6.3.6 A pressure drop test to identify excessive degradation of resilient valve seals shall be conducted on the:

a. Required Containment Purge Supply and Exhaust Isolation Valves in accordance with the Surveillance Frequency Control Program.

b. Deleted.

4.6.3.7 The required containment purge supply and exhaust isolation valves shall be determined closed in accordance with the Surveillance Frequency Control Program.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves (MSSVs) associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-4.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or two main steam line code safety valves inoperable in one or more steam generators, operation in Modes 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or reduce power to less than or equal to the applicable percent of RATED THERMAL POWER per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With three main steam line code safety valves inoperable in one or more steam generators, operation in Modes 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valves are restored to OPERABLE status or reduce power to less than or equal to the applicable percent of RATED THERMAL POWER per Table 3.7-1 and within 36 hours, reduce the Power Range Neutron Flux High trip setpoint to less than or equal to the RATED THERMAL POWER per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Verify each required MSSV lift setpoint per Table 3.7-4. No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

Otherwise, be in MODE 2 within the next 6 hours.

MODES 2 - and 3 With one or more main steam line isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may proceed provided;

a. The isolation valve(s) is (are) maintained closed, and

b. The isolation valve(s) is (are) verified closed once per 7 days.

Otherwise, be in MODE 3, HOT STANDBY, within the next 6 hours, and MODE 4, HOT SHUTDOWN, within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops operable, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per the INSERVICE TESTING PROGRAM.

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* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

6.8.4.j **Deleted**

6.8.4.k **Reactor Coolant Pump Flywheel Inspection Program**

In addition to the requirements of the ISI Program, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 319 AND 300 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 application dated August 30, 2016,[1] PSEG Nuclear LLC (the licensee) requested changes to the Technical Specifications (TSs) for the Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem Units 1 and 2). The TSs are contained in Appendix A of each unit's renewed facility operating license. Specifically, the licensee requested to adopt U.S. Nuclear Regulatory Commission (NRC or the Commission)-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015.[2]

The proposed change requests a revision to the TSs to eliminate TS 6.8.4.j, "Inservice Testing Program." A new defined term, "Inservice Testing Program," is requested to be added to the TSs "Definitions" section. The licensee stated that the license amendment request is consistent with NRC-approved Traveler TSTF-545, Revision 3. This TS improvement was made available by letter to the TSTF dated December 11, 2015,[3] as part of the consolidated line item improvement process (CLIIP). A notice of availability was published in the Federal Register (FR) on March 28, 2016 (81 FR 17208).

The licensee's letter dated August 30, 2016,[4] also included a request to use American Society of Mechanical Engineers (ASME) Code Case OMN-20, "Inservice Test Frequency," as an alternative to certain ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) requirements. The NRC considered this request separately from the proposed license amendments and authorized the licensee's use of this alternative by letter dated May 19, 2017.[5]

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML16243A233
[2] ADAMS Accession No. ML15294A555
[3] ADAMS Package Accession No. ML15317A071
[4] ADAMS Accession No. ML16243A233
[4] ADAMS Accession No. ML17132A005

2.0 REGULATORY EVALUATION

2.1 Background

An inservice test is a test to assess the operational readiness of a structure, system, or component after first electrical generation by nuclear heat. The ASME OM Code provides requirements for inservice testing (IST) of certain components in light-water nuclear power plants. The ASME OM Code identifies the components subject to the testing (i.e., pumps, valves, pressure relief devices, and dynamic restraints), responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating results, corrective actions, personnel qualification, and recordkeeping.

On August 23, 2012, the NRC issued Regulatory Issue Summary 2012-10, "NRC Staff Position on Applying Surveillance Requirements 3.0.2 and 3.0.3 to Administrative Controls Program Tests."[6] The regulatory issue summary states that the NRC staff had determined that restructuring TS chapters during the development of the improved Standard Technical Specifications (STSs) resulted in unintended consequences when SRs 3.0.2 and 3.0.3 provisions were made applicable to the IST program. The NRC staff concluded that SRs 3.0.2 and 3.0.3 cannot be applied to TS Section 5.5 tests that are not associated with an SR.

TSTF-545, Revision 3, describes how to request license amendments that would eliminate conflicting requirements between Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, "Codes and standards," and the TSs. TSTF-545, Revision 3, describes elimination of the IST program from the Administrative Controls section of the TSs. The TSs contain surveillances that require testing or test intervals in accordance with the IST program. TSTF 545, Revision 3, describes adding a new definition, "INSERVICE TESTING PROGRAM," to the TSs, which would be defined as "the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." TSTF-545, Revision 3, describes replacement of existing uses of the term, "Inservice Testing Program," with the defined term, as denoted by capital letters, throughout the TSs.

2.2 Proposed Technical Specification Changes

The licensee requested to delete TS 6.8.4.j from the Administrative Controls section of the Salem Units 1 and 2 TSs and replace it with the word "Deleted." TS 6.8.4.j currently states:

> This Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:
>
> a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days

[5] ADAMS Accession No. ML12079A393

Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days;

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies and to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

TS 6.8.4.j, which references Specification 4.0.2 (SR 4.0.2), allows an extension of IST intervals by up to 25 percent of the specified surveillance interval. If it is discovered that a surveillance associated with an IST activity was not performed within the required interval, Specification 4.0.3 (SR 4.0.3) allows the licensee to delay declaring the associated limiting condition for operation (LCO) not met in order to perform the missed surveillance. The licensee did not request changes to SRs 4.0.2 or 4.0.3.

The licensee requested to revise the Definitions section of the TSs by adding Definition 1.15.1, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The licensee also requested that all existing occurrences of "Inservice Testing Program" in TS SRs be replaced with "INSERVICE TESTING PROGRAM," so that the SRs refer to the new definition in lieu of the deleted program. In addition, the licensee proposed conforming changes to the TS index pages denoting the addition of the new definition.

2.3 Regulatory Requirements and Guidance

The NRC staff considered the following regulatory requirements, guidance, and licensing information during its review of the proposed changes.

As described in 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," whenever a holder of an operating license desires to amend the license, application for an amendment must be filed with the Commission fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. For TSs, 10 CFR 50.36(a)(1) states that each applicant for an operating license shall include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36. Also, 10 CFR 50.36(a)(1) states that a summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TSs.

Pursuant to 10 CFR 50.92(a), in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate. The issuance of operating licenses is addressed by 10 CFR 50.57(a), and requires the Commission to find, among other things, that "[t]here is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter." It also requires a finding that "[t]he issuance of the license will not be inimical to the common defense and security or to the health and safety of the public."

Technical Specifications

Per 10 CFR 50.36(b), each license authorizing operation of a utilization facility will include TSs. The TSs will be derived from the analyses and evaluations included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34 (describing the technical information to be included in applications for an operating license). The Commission may include such additional TSs as the Commission finds appropriate.

Paragraph 50.36(c) of 10 CFR requires TSs to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. Paragraph 50.36(c)(3) of 10 CFR states that "[s]urveillance 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 limiting conditions for operation will be met." Paragraph 50.36(c)(5) of 10 CFR states that "[a]dministrative 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."

The NRC staff's guidance for review of the TSs is in Chapter 16, "Technical Specifications," of NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," dated March 2010.[7] As described therein, as part of the regulatory standardization effort, the NRC staff has prepared improved STSs (NUREG 1430 through NUREG 1434) for each of the LWR nuclear steam supply systems and associated balance-of-plant equipment systems. Accordingly, the NRC staff's review includes consideration of whether the proposed changes are consistent with TSTF-545, Revision 3. Special attention is given to TS provisions that depart from the improved STSs, as modified by NRC-approved TSTF travelers, to determine whether proposed differences are justified by uniqueness in plant design or other considerations so that 10 CFR 50.36 is met. In addition, the guidance states that comparing the change to previous STSs can help clarify the intent of the TSs.

Inservice Testing

Pursuant to 10 CFR 50.54, "Conditions of licenses," the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. The regulation at 10 CFR 50.55a(f) addresses IST and requires that systems and components of boiling and pressurized water-cooled nuclear power reactors meet the requirements of the ASME Boiler and Pressure Vessel (BPV) Code and ASME OM Code as specified in 10 CFR 50.55a(f)(1) through (f)(6).

The ASME OM Code is a consensus standard that is incorporated by reference into 10 CFR 50.55a. During the incorporation process, the NRC staff reviewed the ASME OM Code requirements for technical sufficiency and found that the ASME OM Code IST program requirements were suitable for incorporation into the NRC's rules.

[7] ADAMS Accession No. ML100351425

Since Salem Units 1 and 2 were issued construction permits (CPs) on September 25, 1968, the provisions of 10 CFR 50.55a(f)(1) apply, which state:

> *Inservice testing requirements for older plants (pre-1971 CPs).* For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, pumps and valves must meet the test requirements of paragraphs (f)(4) and (5) of this section to the extent practical. Pumps and valves that are part of the reactor coolant pressure boundary must meet the requirements applicable to components that are classified as ASME Code Class 1. Other pumps and valves that perform a function to shut down the reactor or maintain the reactor in a safe shutdown condition, mitigate the consequences of an accident, or provide overpressure protection for safety-related systems (in meeting the requirements of the 1986 Edition, or later, of the BPV or OM Code) must meet the test requirements applicable to components that are classified as ASME Code Class 2 or Class 3.

The regulation in 10 CFR 50.55(a)(f)(5)(ii) states, in part: "If a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program."

The NRC staff's guidance for functional design, qualification, and IST programs for pumps, valves, and dynamic restraints is in Section 3.9.6 of NUREG-0800, Revision 3, dated March 2007.[8] As part of the review for the IST program for pumps, the NRC staff will review the IST frequencies and test parameters. The frequency of ISTs and test parameters are acceptable if the provisions of Subsection ISTB-3000 of the ASME OM Code are met. As described therein, the licensee's IST program is acceptable if the program meets the requirements of the ASME Code, Section XI, or the ASME OM Code, as incorporated by reference in 10 CFR 50.55a.

The NRC staff's guidance for complying with the codes and standards in 10 CFR 50.55a is in Section 5.2.1.1 of NUREG-0800, Revision 3, dated March 2007.[9] As stated therein, acceptance criteria are based in part on meeting 10 CFR 50.55a.

The NRC staff's guidance for review of inservice inspection and testing of Class 2 and Class 3 components is in Section 6.6 of NUREG-0800, Revision 2, dated March 2007.[10] As stated therein, acceptance criteria are based on meeting the relevant parts of 10 CFR 50.55a as they pertain to the specification of the preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and Class 3 systems and components.

NUREG-1482, Revision 2, "Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants, Final Report," dated October 2013,[11] provides guidance for the inservice testing of pumps and valves.

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[8] ADAMS Accession No. ML070720041
[9] ADAMS Accession No. ML070040003
[10] ADAMS Accession No. ML070550071
[11] ADAMS Accession No. ML13295A020

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine whether the proposed changes are consistent with the regulations, guidance, and licensing information discussed in Section 2.3 of this safety evaluation. In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Among the considerations is whether the TSs, as amended, would provide the necessary administrative controls per 10 CFR 50.36(c)(5).

In making its determination as to whether to amend the licenses, the NRC staff considered those regulatory requirements that are automatically conditions of the licenses pursuant to 10 CFR 50.54. Where the regulations already condition the licenses, there is no need for a duplicative requirement in the TSs; the regulations provide the necessary reasonable assurance of the health and safety of the public.

3.1 Assessment of Requested Deletion of TS 6.8.4.j, "Inservice Testing Program"

The Salem Units 1 and 2 TS 6.8.4.j, which is in the Administrative Controls section, requires the licensee to have an IST program that provides controls for IST of ASME Code Class 1, 2, and 3 components. The NRC staff notes that the licensee's IST, which is required by 10 CFR 50.54 and 50.55a(f), and which is outside the scope of this amendment request, already contains requirements and considerations similar to those of TS 6.8.4.j. Therefore, requiring the licensee to have an IST program in TSs is duplicative of the license condition in 10 CFR 50.54. Thus, with the proposed TS changes, the licensee will still be required to maintain an IST program in accordance with the ASME OM Code, as specified in 10 CFR 50.55a(f). For the reasons explained further in this safety evaluation, it is not necessary to have additional administrative controls in the TSs for Salem Units 1 and 2 relating to the IST program to assure operation of the facility in a safe manner.

Consideration of TS 6.8.4.j.a

The ASME OM Code requires testing to normally be performed within certain time periods. TS 6.8.4.j.a more precisely defines those time periods specified in the ASME OM Code and applicable addenda (e.g., states "at least once per 31 days" for the ASME OM Code frequency of "monthly"). However, the NRC staff has determined that the ASME OM Code frequencies are sufficient to assure operation of the facility in a safe manner. Therefore, the more precise definitions in TS 6.8.4.j.a are not necessary to assure operation of the facility in a safe manner.

Consideration of TS 6.8.4.j.b

TS 6.8.4.j.b allows the licensee to extend, by up to 25 percent, the interval between IST activities, as required by TS 6.8.4.j.a, and for other normal and accelerated frequencies specified as 2 years or less in the IST program. Similar to TS 6.8.4.j.b, the NRC authorization of ASME Code Case OMN 20, by letter dated May 19, 2017,[12] also permits the licensee to extend the IST intervals specified in the ASME OM Code by up to 25 percent.

The NRC staff determined that the TS 6.8.4.j.b allowance to extend IST intervals is not needed to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that deletion of TS 6.8.4.j.b is acceptable. Moreover, the deletion of TS 6.8.4.j.b does not impact the

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[12] ADAMS Accession No. ML17132A005

licensee's ability to extend IST intervals using ASME Code Case OMN-20, as authorized by the NRC.

Consideration of TS 6.8.4.j.c

TS 6.8.4.j.c allows the licensee to use Specification 4.0.3 when it discovers that an SR associated with an inservice test was not performed within its specified frequency. Specification 4.0.3 allows the licensee to delay declaring an LCO not met in order to perform the missed surveillance. The use of Specification 4.0.3 for inservice tests is limited to those inservice tests required by an SR. In accordance with 10 CFR 50.55a, the licensee may also request relief from the ASME OM Code requirements to address issues associated with a missed inservice test. The deletion of TS 6.8.4.j.c does not change any of these requirements, and Specification 4.0.3 will continue to apply to those inservice tests required by SRs. Based on the above, the NRC staff determined that deletion of TS 6.8.4.j.c is acceptable.

Consideration of TS 6.8.4.j.d

TS 6.8.4.j.d states that nothing in the ASME OM Code shall be construed to supersede the requirements of any TSs. However, the regulations in 10 CFR 50.55a(f)(5)(ii) address what to do if a revised IST program for a facility conflicts with the TSs for the facility. The regulations require the licensee to apply for an amendment to the TSs to conform the TSs to the revised program at least 6 months prior to the start of the period for which the provisions become applicable. Accordingly, there is no need for a TS stating how to address conflicts between the TSs and the IST program because the regulations specify how conflicts must be resolved.

Conclusion Regarding Deletion of TS 6.8.4.j

As explained above, the NRC staff determined that the requirements currently in TS 6.8.4.j are not necessary to assure operation of the facility in a safe manner. Based on this evaluation, the staff concludes that deletion of TS 6.8.4.j from the licensee's TSs is acceptable because TS 6.8.4.j. is not required by 10 CFR 50.36(c)(5).

3.2 Definition of INSERVICE TESTING PROGRAM and Revision to Surveillance Requirements

The licensee proposed to revise the TS Definitions section to include the term, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The proposed definition is consistent with the definition in the NRC-approved TSTF-545, Revision 3. The NRC staff finds the definition acceptable because it correctly refers to the IST requirements in 10 CFR 50.55a(f).

The licensee requested that all existing references to the "Inservice Testing Program" in SRs be revised to "INSERVICE TESTING PROGRAM" to reference the new TS-defined term in lieu of the deleted IST program TSs. The proposed change is consistent with the intent of TSTF-545, Revision 3, to replace the current references in SRs with the new definition. The NRC staff verified that for each SR reference to the "Inservice Testing Program," the licensee proposed to change the reference to "INSERVICE TESTING PROGRAM." The proposed change does not alter how the SR testing is performed. The IST frequencies could change because the TSs will no longer include the more precise test frequencies in TS 6.8.4.j.a. However, as discussed in Section 3.1 of this safety evaluation, the staff determined that the TSs do not need to include the more precise test frequencies currently in TS 6.8.4.j.a in order to assure operation of the facility in a safe manner.

Based on its review, the NRC staff determined that revising the SRs to refer to the new definition is acceptable because these SRs will continue to be performed in accordance with the requirements of 10 CFR 50.55a(f). The staff also determined that, with the proposed changes, 10 CFR 50.36(c)(3) will continue to be met because the SRs will continue 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.

3.3 Conforming Changes and Variations from TSTF-545, Revision 3

The NRC staff also evaluated the following conforming changes and variations from TSTF-545, Revision 3, not previously addressed in this safety evaluation.

a. The Salem Units 1 and 2 TSs have not been converted to the improved STSs on which TSTF-545, Revision 3, is based. As a result, the numbering, format, and content of the Salem Units 1 and 2 TSs vary from TSTF-545, Revision 3. In addition, the Salem Units 1 and 2 TSs use different numbering than the improved STSs on which TSTF-545, Revision 3, is based. The NRC staff finds that the licensee's proposed deviations in numbering, format, and content are editorial in nature and that the licensee's proposed TS changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds the licensee's proposed TS changes acceptable.

b. An index is included in the Salem Unit 1 and 2 TSs. Therefore, the licensee included conforming changes to the index resulting from the addition of the new definition. The NRC staff finds that the licensee's proposed deviations are editorial in nature and consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds the licensee's proposed TS changes acceptable.

c. The licensee proposed to replace the content of the Salem, Unit 1 and 2 IST TSs with the word "Deleted" and retain the existing numbering sequence. The NRC staff finds that these proposed changes are editorial in nature and consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds the licensee's proposed TS changes acceptable.

The NRC staff finds that the proposed changes and variations from TSTF-545, Revision 3, are editorial in nature, and that the licensee's proposed TS changes remain consistent with the intent of TSTF-545, Revision 3. Therefore, the NRC staff finds that the licensee's proposed TS changes are 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 30, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding on November 8, 2016 (81 FR 78651). 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.

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