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

August 20, 2010

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: SURVEILLANCE REQUIREMENTS FOR INSERVICE INSPECTION AND INSERVICE TESTING (TAC NOS. ME2277 AND ME2278)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment Nos. 297 and 279 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) and Facility Operating Licenses (FOLs) in response to your application dated September 23, 2009, as supplemented by letter dated December 21, 2009.

The amendments revise the TSs to: (1) delete TS 4.0.5, which pertains to surveillance requirements for inservice inspection (ISI) and inservice testing (IST) of American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* Class 1, 2 and 3 components; (2) add a new TS for the IST Program to Section 6.0, "Administrative Controls," of the TSs; and (3) change TSs that currently reference TS 4.0.5 to reference the IST Program or ISI Program, as applicable.

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

Sincerely,

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 297 to License No. DPR-70
2. Amendment No. 279 to License No. DPR-75
3. Safety Evaluation

cc w/encls: 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 FACILITY OPERATING LICENSE

Amendment No. 297
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 September 23, 2009, as supplemented by letter dated December 21, 2009, 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 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 Appendices A and B, as revised through Amendment No. 297, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 20, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 297

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following page of 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 4	Page 4

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
3/4 0-2a	- - -
3/4 0-3	3/4 0-3
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 4-32	3/4 4-32
3/4 4-33	3/4 4-33
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 7-28	3/4 7-28
3/4 9-8a	3/4 9-8a
- - -	6-19e
6-26	6-26

(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

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 297, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(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 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 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 Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:

> PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified frequency shall be failure to meet the Limiting Condition for Operation, except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

4.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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% $\Delta k/k$ 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.

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%.

3/4.4.2 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 at least once per 18 months 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 at least once per 92 days 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. At least once per 18 months.

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 at least once per 31 days thereafter when the POPS is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel at least once per 18 months.

c. Verifying the POPS isolation valve is open at least once per 72 hours 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 at least once per 12 hours* when the vents(s) is being used for overpressure protection.

* 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 at least once per 31 days.

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of the Inservice Inspection Program, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the requirements of the Inservice Inspection 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.

4.4.10.1.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads - The steam generator channel heads shall be ultrasonically inspected during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material. The stainless steel clad surfaces of the steam generator channel heads shall also be visually inspected during the above outages. This may be accomplished by direct visual examination or by remote means such as television camera. If the visual examination, either direct or remote, reveals detectable cladding indications, a record shall be made by means of a video tape recording or photographs for comparison purposes.

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. At least once per 18 months.

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.

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. At least once per 31 days 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. At least once per 18 months 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 at least once per 18 months 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 ≤ 60% opening angle.

4.6.3.1.3 At least once per 18 months, 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 at least once per 72 hours, 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 at least once per 6 months.

b. Deleted.

4.6.3.1.7 The required containment purge supply and exhaust isolation valves shall be determined closed at least once per 31 days.

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 COLD SHUTDOWN within the following 30 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 COLD SHUTDOWN within the following 30 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.

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours, replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9c on the supported component or declare the supported system inoperable and follow appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program, in addition to the regular Inservice Inspection Program requirements.

a. Visual Inspection

All snubbers shall be categorized into two groups: those accessible and those inaccessible during reactor operation. The visual inspection interval for each category of snubbers shall be determined based upon the criteria provided in Table 4.7-3.

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.

6.8.4.j Inservice Testing Program

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
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.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections.

i. Records of Quality Assurance activities required by the QA Manual.

j. DELETED

k. Records of SORC meetings and activities of the Nuclear Review Board (and activities of its predecessor, the Offsite Safety Review (OSR) staff.)

l. Records for Environmental Qualification which are covered under the provisions of Paragraph 6.16.

m. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

n. Records of secondary water sampling and water quality.

o. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

p. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

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 FACILITY OPERATING LICENSE

Amendment No. 279
License No. DPR-75

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 September 23, 2009, as supplemented by letter dated December 21, 2009, 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 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 Appendices A and B, as revised through Amendment No. 279, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 20, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 279

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

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

Remove	Insert
Page 4	Page 4

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
3/4 0-2a	- - -
3/4 0-3	3/4 0-3
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 4-33	3/4 4-33
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 7-23	3/4 7-23
3/4 9-9	3/4 9-9
6-19f	6-19f
6-26	6-26

(2) Technical Specifications and Environmental Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 279, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified frequency shall be failure to meet the Limiting Condition for Operation, except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

4.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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%.

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%.

3/4.4.5 RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.5.1 In addition to the requirements of the Inservice Testing Program, each PORV shall be demonstrated OPERABLE at least once per 18 months 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 at least once per 92 days 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 at least once per 72 hours during steady state operation,

d*. Performance of a Reactor Coolant System water inventory balance** at least once per 72 hours. 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 at least once per 24 hours.

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. At least once per 18 months.

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 at least once per 31 days thereafter when the POPS is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel at least once per 18 months.

c. Verifying the POPS isolation valve is open at least once per 72 hours 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 at least once per 12 hours* when the vent(s) is being used for overpressure protection.

* 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 at least once per 31 days.

REACTOR COOLANT SYSTEM

3.4.11 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.11.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.11.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of the Inservice Inspection 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.

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. At least once per 18 months.

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. At least once per 31 days 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. At least once per 18 months 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 at least once per 18 months 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 ≤ 60° opening angle.

4.6.3.3 At least once per 18 months, 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 at least once per 72 hours, 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 at least once per 6 months.

b. Deleted.

4.6.3.7 The required containment purge supply and exhaust isolation valves shall be determined closed at least once per 31 days.

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 COLD SHUTDOWN within the following 30 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 COLD SHUTDOWN within the following 30 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.

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours, replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9c on the supported component or declare the supported system inoperable and follow appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the regular Inservice Inspection Program requirements.

a. Visual Inspection

All snubbers shall be categorized into two groups: those accessible and those inaccessible during reactor operation. The visual inspection interval for each category of snubbers shall be determined based upon the criteria provided in Table 4.7-3.

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.

4. When the W* methodology has been implemented, inspect 100 percent of the inservice tubes for the entire hot-leg tubesheet W* distance with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 6.8.4.i.c.1 every 24 effective full power months or one refueling outage (whichever is less).

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

6.8.4.j Inservice Testing Program

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
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.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections.

i. Records of Quality Assurance activities required by the QA Manual.

j. DELETED

k. Records of SORC meetings and activities of the Nuclear Review Board (and activities of its predecessor, the Offsite Safety Review (OSR) staff).

l. Records for Environmental Qualification which are covered under the provisions of Paragraph 2.C(7) and 2.C(8) of Facility Operating License DPR-75.

m. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

n. Records of secondary water sampling and water quality.

o. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

p. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 297 AND 279 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PSEG NUCLEAR, LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated September 23, 2009, as supplemented by letter dated December 21, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML092790463 and ML100040004, respectively), PSEG Nuclear, LLC (PSEG or the licensee) submitted a request for changes to the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2, Technical Specifications (TSs) and Facility Operating Licenses (FOLs).

The proposed amendment would revise the TSs to: (1) delete TS 4.0.5, which pertains to surveillance requirements (SRs) for inservice inspection (ISI) and inservice testing (IST) of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) Class 1, 2 and 3 components; (2) add a new TS for the IST Program to Section 6.0, "Administrative Controls," of the TSs; and (3) change TSs that currently reference TS 4.0.5 to reference the IST Program or ISI Program, as applicable. The new TS for the IST Program, TS 6.8.4.j, will indicate that the program will include testing frequencies applicable to the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), replacing the current reference to Section XI of the ASME Code specified in TS 4.0.5. In addition, TS 6.8.4.j would revise the requirements, currently contained in TS 4.0.5, regarding the applicability of the surveillance interval extension provisions of SR 4.0.2.

The licensee's application dated September 23, 2009, stated that the license amendment request proposes changes for consistency with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55(a)(f)(4) for IST of pumps and valves and removes TS requirements that are redundant to the requirements of 10 CFR 50.55a, "Codes and standards." The licensee's application also stated that the proposed changes are consistent with Nuclear Regulatory Commission (NRC or the Commission)-approved Technical Specification Task Force (TSTF) Travelers TSTF-479, "Changes to Reflect Revision to 10 CFR 50.55a," and TSTF-497, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less; and NUREG-1431, Revision 3.0, "Standard Technical Specifications - Westinghouse Plants."

Enclosure

The supplement dated December 21, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 29, 2009 (74 FR 68871).

2.0 REGULATORY EVALUATION

The Commission's regulatory requirements related to the content of the TSs are set forth in 10 CFR 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; (3) SRs; (4) design features; and (5) administrative controls. Paragraph (c)(3) of 10 CFR 50.36 states that SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulations in 10 CFR 50.55a(f)(4) and (g)(4) establish the ASME Code edition and addenda to be used by licensees for performing IST of pumps and valves and ISI of components (including supports). Paragraphs 50.55a(f)(4)(ii) and (g)(4)(ii) require the use of the latest edition and addenda that has been incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the beginning of each 120-month interval. With respect to the requirements for IST, the ASME OM Code was initially incorporated by reference in 10 CFR 50.55a(b) in a final rule dated September 22, 1999 (64 FR 51370). Prior to the final rule, IST programs were required to meet the requirements of Section XI of the ASME Code.

Paragraph (f)(5)(ii) of 10 CFR 50.55a requires that, if a revised IST program for a facility conflicts with the TSs for that facility, the licensee shall apply to the Commission for amendment of the TSs to conform the TSs to the revised program. As discussed in a PSEG letter dated December 31, 2008, "Submittal of Relief Requests Associated with the Fourth Inservice Testing Interval" (ADAMS Accession No. ML090130525), the fourth 10-year interval of the IST program at Salem Units 1 and 2, which began on August 31, 2009, was developed in accordance with the 2001 Edition through the 2003 Addenda of the OM Code. Salem Unit 1 and 2 TS 4.0.5 currently references Section XI of the ASME Code for IST requirements. As such, a revision to the TSs is needed in accordance with 10 CFR 50.55a(f)(5)(ii) to conform the TSs to the revised program.

NUREG-1431, Revision 3.0, "Standard Technical Specifications - Westinghouse Plants," dated June 2004 (ADAMS Accession No. ML041830612), contains the improved Standard Technical Specifications (STS) for Westinghouse plants.

TSTF-479, Revision 0, "Changes to Reflect Revisions of 10 CFR 50.55a," dated December 2, 2004 (ADAMS Accession No. ML052990317), proposed changes to the improved STS, NUREG-1430 through 1434, to reflect the current edition of the ASME Code specified in 10 CFR 50.55a(b) (i.e., ASME OM Code rather than Section XI of the ASME Code). The NRC staff approved TSTF-479, Revision 0, in a letter dated December 6, 2005 (ADAMS Accession No. ML053460302).

TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less," dated July 12, 2006 (ADAMS Accession No. ML061930221), proposed changes to the improved STS, NUREG-1430 through 1434, to revise the STS section regarding the IST program by clarifying that the application of the 25% IST interval extension allowed by STS SR 3.0.2 was for IST frequencies of 2 years or less. The NRC staff approved TSTF-497, Revision 0, in a letter dated October 4, 2006 (ADAMS Accession No. ML062780321).

NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," dated January 2005 (ADAMS Accession No. ML050550290), provides guidelines and recommendations for developing and implementing programs for the IST of pumps and valves at commercial nuclear power plants.

3.0 TECHNICAL EVALUATION

3.1 Specific Proposed TS Changes

As discussed above in Safety Evaluation (SE) Section 1.0, the proposed amendment would delete TS 4.0.5 which pertains to SRs for ISI and IST of ASME Code Class 1, 2 and 3 components. The IST portion of TS 4.0.5 would be relocated to new TS 6.8.4.j, "Inservice Testing Program," consistent with NUREG-1431, TS 5.5.8, "Inservice Testing Program." Consistent with TSTF-479, the new TS for the IST Program will indicate that the program will include testing frequencies applicable to the ASME OM Code, replacing the current reference to Section XI of the ASME Code specified in TS 4.0.5. In addition, TS 6.8.4.j would revise the requirements, currently contained in TS 4.0.5, regarding the applicability of the surveillance interval extension provisions of SR 4.0.2 for consistency with TSTF-497. The ISI portion of TS 4.0.5 would be removed from the Salem TSs consistent with NUREG-1431.

The new TS 6.8.4.j would read as follows:

6.8.4.j INSERVICE TESTING PROGRAM

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
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once 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 two 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.

The following Salem Unit 1 TSs would be revised to remove references to TS 4.0.5 and to add references to either the IST Program or the ISI Program, as applicable:

SR 4.1.2.3	Charging Pump - Shutdown
SR 4.1.2.4	Charging Pumps - Operating
SR 4.4.2.1	Safety Valves - Shutdown
SR 4.4.2.2	Safety Valves - Operating
SR 4.4.3.1	Relief Valves
SR 4.4.6.3	Primary Coolant System Pressure Isolation Valves
SR 4.4.9.3.1	Overpressure Protection Systems
SR 4.4.10.1.1	Structural Integrity - ASME Code Class 1, 2 and 3 Components
SR 4.5.2	Emergency Core Cooling Systems
SR 4.6.2.1	Containment Spray System
SR 4.6.3.1.4	Containment Isolation Valves
SR 4.7.1.1	Turbine Cycle Safety Valves
SR 4.7.1.5	Main Steam Line Isolation Valves
SR 4.7.9	Snubbers
SR 4.9.8.2	Low Water Level
TS 6.10	Record Retention

The following Salem Unit 2 TSs would be revised to remove references to TS 4.0.5 and to add references to either the IST Program or the ISI Program, as applicable:

SR 4.1.2.3	Charging Pump - Shutdown
SR 4.1.2.4	Charging Pumps - Operating
SR 4.4.2.1	Safety Valves - Shutdown
SR 4.4.3	Safety Valves - Operating
SR 4.4.5.1	Relief Valves
SR 4.4.7.2.2	Leakage Detection Systems
SR 4.4.10.3.1	Overpressure Protection Systems
SR 4.4.11.1	Structural Integrity - ASME Code Class 1, 2 and 3 Components
SR 4.5.2	Emergency Core Cooling Systems
SR 4.6.2.1	Containment Spray System
SR 4.6.3.4	Containment Isolation Valves
SR 4.7.1.1	Turbine Cycle Safety Valves
SR 4.7.1.5	Main Steam Line Isolation Valves
SR 4.7.9	Snubbers
SR 4.9.8.2	Low Water Level
TS 6.10	Record Retention

3.2 Evaluation of Proposed Changes Related to IST Program

In 1990, the ASME published the initial edition of the OM Code, which provides rules for IST of pumps and valves. The OM Code was developed and is maintained by the ASME Committee on Operation and Maintenance of Nuclear Power Plants. The OM Code was developed in response to the ASME Board on Nuclear Codes and Standards directive that transferred responsibility for development and maintenance of rules for the IST of pumps and valves from the ASME Code, Section XI, Subcommittee on Nuclear Inservice Inspection to the ASME OM Committee. The ASME intended the OM Code to replace Section XI rules for IST of pumps and valves, and the rules for IST of pumps and valves have been deleted from Section XI of the ASME Code.

Section 50.55a(f) of 10 CFR, "Inservice Testing Requirements," requires, in part, that ASME Code Class 1, 2, and 3 pumps and valves meet the testing requirements of the OM Code. The Salem Units 1 and 2 fourth 10-year interval IST program was updated to comply with the 2001 Edition through the 2003 Addenda of the OM Code as required by 10 CFR 50.55a(f)(4)(ii). As a consequence, the reference in TS 4.0.5 to Section XI of the ASME Code for IST requirements results in a reference to a deleted portion of the ASME Code, and a revision to the TS is required. The proposed amendment was submitted, in part, to revise the TSs to reference the current OM Code requirements.

The proposed new TS 6.8.4.j would replace the IST portion of the TS 4.0.5. The proposed TS references the OM Code applicable to the Salem IST program and is consistent with 10 CFR 50.55a(f)(4)(ii). The proposed changes do not eliminate any inservice tests and do not relieve the licensee of its responsibility to seek relief from Code test requirements when the licensee determines that conformance with the requirements is impractical. The proposed changes will eliminate the inconsistency between the TSs for Salem Units 1 and 2 and the Salem IST program as required by 10 CFR 50.55a(f)(5)(ii). Therefore, the NRC staff finds the proposed changes, related to referencing the OM Code in lieu of the Section XI of the ASME Code, to be acceptable. The change is also consistent with NRC-approved TSTF-479.

The proposed amendment would allow the application of the 25% IST interval extension, provided for in SR 4.0.2, to specified normal and accelerated SR frequencies in the TS, but will limit the applicability to test intervals of 2 years or less. This is consistent with the intent that the extension would provide operational flexibility, but would not significantly degrade the reliability that results from performing the surveillance at a specified frequency. Further, the proposal to limit the applicability to frequencies of 2 years or less limits the maximum incremental time period, between surveillances, which could be added by the 25% extension. Without this limitation, some components, such as safety and relief valves, which may be tested at surveillance intervals greater than 2 years, could have extensions applied which would be much greater than needed for operational flexibility. Based on the above considerations, the NRC staff finds the proposed changes, related to IST interval extensions, to be acceptable. The change is also consistent with NRC-approved TSTF-497 and the guidance in NUREG-1482.

3.3 Evaluation of Proposed Changes Related to ISI Program

The NRC's requirements in 10 CFR 50.55a(g) state, in part, that ASME Code Class 1, 2, and 3 components and their supports must meet the requirements of the ASME Code. The ASME publishes a new edition of the ASME Code every 3 years, and a new addendum every year. The third 10-year ISI interval at Salem Units 1 and 2 was updated to comply with the 1998 Edition through 2000 Addenda to the ASME Code, as required by 10 CFR 50.55a(g)(4). For Salem Unit 1, the third interval began on May 19, 2001, and will end on May 20, 2011. For Salem Unit 2, the third interval began on November 27, 2003, and will end on November 27, 2013.

Per the current requirements in TS 4.0.5, the ISI of ASME Code Class 1, 2, and 3 components shall be performed in accordance with applicable editions and addenda of the ASME Code, Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

The NRC staff finds that the removal of the ISI requirements, currently in TS SR 4.0.5, does not eliminate any ISI not covered by 10 CFR 50.55a, and does not relieve the licensee of its responsibility to seek relief from the ASME Code requirements when they are impractical. The proposed changes will eliminate the regulatory redundancy between the TSs and 10 CFR 50.55a. Based on the above considerations, the NRC staff finds the proposed changes related to removal of the ISI program requirements, currently contained in TS 4.0.5, to be acceptable. The change is also consistent with NUREG-1431.

3.4 Evaluation of Other TS Changes

As discussed above in SE Section 3.1, the proposed amendment would revise a number of TSs to remove references to TS 4.0.5 and to add references to either the IST program or the ISI program, as applicable. Due to the proposed deletion of TS 4.0.5, these other TS changes are considered administrative in nature. Therefore, the NRC staff finds these changes to be acceptable.

3.5 Technical Evaluation Conclusion

Based on the considerations in SE Sections 3.2, 3.3 and 3.4, the NRC staff concludes that the proposed amendment is 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. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (74 FR 68871). 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) 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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Date: August 20, 2010

August 20, 2010

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: SURVEILLANCE REQUIREMENTS FOR INSERVICE INSPECTION AND INSERVICE TESTING (TAC NOS. ME2277 AND ME2278)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment Nos. 297 and 279 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) and Facility Operating Licenses (FOLs) in response to your application dated September 23, 2009, as supplemented by letter dated December 21, 2009.

The amendments revise the TSs to: (1) delete TS 4.0.5, which pertains to surveillance requirements for inservice inspection (ISI) and inservice testing (IST) of American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* Class 1, 2 and 3 components; (2) add a new TS for the IST Program to Section 6.0, "Administrative Controls," of the TSs; and (3) change TSs that currently reference TS 4.0.5 to reference the IST Program or ISI Program, as applicable.

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

Sincerely,
/ra/
Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311
Enclosures:
1. Amendment No. 297 to License No. DPR-70
2. Amendment No. 279 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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