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

June 15, 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: MISCELLANEOUS ADMINISTRATIVE AND EDITORIAL CHANGES (TAC NOS. ME2229 AND ME2230)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment Nos. 295 and 278 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 14, 2009, as supplemented by letter dated April 12, 2010.

The amendments make miscellaneous administrative and editorial changes to the TS and FOLs including correction of typographical and format errors, correction of administrative differences between units, and deletion of historical requirements that have expired.

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. 295 to License No. DPR-70
2. Amendment No. 278 to License No. DPR-75
3. Safety Evaluation

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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 FACILITY OPERATING LICENSE

Amendment No. 295
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 14, 2009, as supplemented by letter dated April 12, 2010, 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. 295, 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: June 15, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 295

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following pages of Facility Operating License No. DPR-70 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
Page 4	Page 4
Page 5c	Page 5c
Attachment 1, Pages 1-4	- - -

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 1-9	3/4 1-9
3/4 1-18a	3/4 1-18a
3/4 1-19	3/4 1-19
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 2-7	3/4 2-7
3/4 2-14	3/4 2-14
3/4 3-13	3/4 3-13
3/4 3-31a	3/4 3-31a
3/4 3-36a	3/4 3-36a
3/4 3-38a	3/4 3-38a
3/4 3-57a	3/4 3-57a
3/4 3-71	3/4 3-71
3/4 5-2	3/4 5-2
3/4 5-5	3/4 5-5
3/4 7-21	3/4 7-21
3/4 7-34	3/4 7-34
3/4 8-9a	3/4 8-9a
3/4 8-14	3/4 8-14

(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. 295, 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.

10. TERMINATION

Pursuant to the provisions of 10 CFR 75.41, the Commission will inform the licensee, in writing, when its installation is no longer subject to Article 39(b) of the principal text of the US/IAEA Safeguards Agreement. The IAEA Safeguards License Conditions incorporating Code 7. of the Facility Attachment as part of NRC License DPR-70 will be terminated as of the date of such notice from the Commission. However, since the IAEA may elect to maintain the licensee's installation under Article 2(a) of the Protocol, provisions equivalent to Codes 1. through 6. of the Facility Attachment (with possible appropriate modifications) may still apply, and accordingly all other IAEA Safeguards License Conditions to NRC License No. DPR-70 will remain in effect until the Commission notifies the licensee otherwise. If this option is not selected by the IAEA, the Commission will then notify the licensee that all License Conditions pertaining to the US/IAEA Safeguards Agreement are terminated.

J. RELOCATED TECHNICAL SPECIFICATIONS

PSEG Nuclear LLC shall relocate certain technical specification requirements to licensee-controlled documents as described below. The location of these requirements shall be retained by the licensee.

a. This license condition approves the relocation of certain technical specification requirements to licensee-controlled documents (UFSAR), as described in the licensee's applications with the staff's safety evaluation approval and Amendment No. as noted below:

Licensee's Applications	Safety Evaluations	Amendment Nos.
September 25, 1996	January 30, 1997	189

Implementation shall include the relocation of technical specifications requirements to the appropriate licensee-controlled document as identified in the licensee's application.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:
1. DELETED
2. Page Changes to Technical Specifications, Appendix A

Date of Issuance: December 1, 1976

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

d. At least once per 18 months by verifying that the flow path required by specification 3.1.2.2.a delivers at least 33 gpm to the Reactor Coolant System.

LIMITING CONDITION FOR OPERATION (Continued)

a) A reevaluation of each accident analysis of table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

c) A core power distribution measurement is obtained and $F_Q(Z)$ $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.c.3.a and 3.1.3.1.c.3.c above are demonstrated.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the limits established in the limiting condition for operation at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2.1 The shutdown and control rod position indication systems shall be OPERABLE and capable of determining the actual and demanded rod positions as follows:

a. Analog rod position indicators, within one hour after rod motion (allowance for thermal soak);

All Shutdown Banks: ± 18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-230 steps.

Control Bank A: ± 18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-230 steps.

Control Bank B: ± 18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 160-230 steps.

Control Bank C and D: ± 18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-230 steps.

b. Group demand counters; ± 2 steps of the pulsed output of the Slave Cycler Circuit over the withdrawal range of 0-230 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one analog rod position indicator per bank inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% RTP or the power distribution monitoring system is inoperable) at least once per 8 hours and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With two or more analog rod position indicators per bank inoperable, within one hour restore the inoperable rod position indicator(s) to OPERABLE status or be in HOT STANDBY within the next 6 hours. A maximum of one rod position indicator per bank may remain inoperable following the hour, with Action (a) above being applicable from the original entry time into the LCO.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 230 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 & 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

APPLICABILITY: MODES 1*, and 2*#@

ACTION:

With a maximum of one shutdown rod not FULLY WITHDRAWN, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. FULLY WITHDRAW the rod, or,

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators**:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

* See Special Test Exceptions 3.10.2 and 3.10.3

** For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

\# With Keff greater than or equal to 1.0

@ Surveillance 4.1.3.4.a is applicable prior to withdrawing control banks in preparation for startup (Mode 2).

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional core power distribution measurements shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the COLR per specification 6.9.1.9.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15% inclusive.

2. Upper core region from 85 to 100% inclusive.

3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2%, and 74.9 ± 2% inclusive.

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
	4 Loops In Operation
Reactor Coolant System T_{avg}	≤ 582.9°F
Pressurizer Pressure	≥ 2200 psia*
Reactor Coolant System Flow	≥ 341,000 gpm#

* Limit not applicable during either THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

\# Includes a 2.4% flow measurement uncertainty plus a 0.1% measurement uncertainty due to feedwater venturi fouling.

TABLE 4.3-1 (Continued)

NOTATION

* With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ 3 percent.

(4) - Manual SSPS functional input check every 18 months.

(5) - Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Deleted

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip mechanism for the Manual Reactor Trip Function.

The Test shall also verify OPERABILITY of the Bypass Breaker Trip circuits.

(10) - DELETED

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Reactor Trip Breaker Undervoltage and Shunt Trip mechanisms.

(12) - DELETED

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	R	1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3,4
c. Containment Pressure-High	S	R	Q(3)	1,2,3
d. Pressurizer Pressure--Low	S	R	Q	1,2,3
e. Differential Pressure Between Steam Lines--High	S	R	Q	1,2,3
f. Steam Flow in Two Steam Lines--High coincident with Tavg--Low-Low or Steam Line Pressure-Low	S	R	Q	1,2,3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	R	1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3,4
c. Containment Pressure--High-High	S	R	Q(3)	1,2,3

TABLE 3.3-6 (Continued)
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 3.0\times10^{-2}$ μCi/cm^3 (Alarm only)	10^{-3}-10^{1} μCi/cm^3	23
2) High Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 1.0\times10^{2}$ μCi/cm^3	10^{-1}-10^{5} μCi/cm^3 (Alarm only)	23
3) Condenser Exhaust System	1	1,2,3&4	$\leq 1.27\times10^{4}$ cpm (Alarm only)	1-10^{6} cpm	23
3. CONTROL ROOM					
a. Air Intake - Radiation Level	2/Intake##	**	$\leq 2.48\times10^{3}$ cpm	10^{1}-10^{7} cpm	24, 25

Control Room air intakes shared between Unit 1 and 2.

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

TABLE 4.3-3 (Continued)
RADIATION MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
2) High Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
3) Condenser Exh. Sys.	S	M	R	Q	1, 2, 3 & 4
3. CONTROL ROOM					
a. Air Intake - Radiation Level	S	M	R	Q	**

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

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

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

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

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILTY: MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.14.1 The operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, and 3.3.3.14.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.14.2 Calibration of the PDMS is required:

a. At least once every 180 Effective Full Power Days when the minimum number and core coverage criteria as defined in 3.3.3.14.b.1 and 3.3.3.14.b.2 are satisfied, or

b. At least once every 31 Effective Full Power Days when only the minimum number criterion as defined in 3.3.3.14.b.3 is satisfied.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of ≥ 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is greater than 1000 psig by verifying that the power lockout switch is in lockout.

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically upon receipt of a safety injection test signal.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. At least once daily (24 hour consecutive period) the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

   1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. 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 safety injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that on a safety injection test signal or control room intake high radiation test signal, the system automatically actuates in the pressurization mode by opening the outside air supply and diverting air flow through the HEPA filter and charcoal adsorber bank.

3. Deleted.

4. Verifying that on a manual actuation signal, the system will actuate to the required pressurization or recirculation operating mode.

5. Verify each CREACS train has the capability to remove the assumed heat load.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

4.7.6.2 Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Habitability Program (Refer to TS 6.18).

LIMITING CONDITION FOR OPERATION

ACTION: MODES 5 and 6 or during movement of irradiated fuel assemblies. *

a. With one chiller inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Restore the chiller to OPERABLE status within 14 days or;

3. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

b. With two chillers inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 2 train within 4 hours and;

3. Restore at least one chiller to OPERABLE status within 72 hours or;

4. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

c. With one chilled water pump inoperable, restore the chilled water pump to OPERABLE status within 7 days or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.10 The chilled water loop which services the safety-related loads in the Auxiliary Building shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each manual valve in the chilled water system flow path servicing safety related components that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months, by verifying that each automatic valve actuates to its correct position on a Safeguards Initiation signal.

c. At least once per 92 days by verifying that each chiller starts and runs.

* During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components are not considered to be inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable.

3. The connection resistance is:

   ≤ 150 micro ohms for inter-cell connections,
   ≤ 350 micro ohms for inter-rack connections,
   ≤ 350 micro ohms for inter-tier connections,
   ≤ 70 micro ohms for field cable terminal connections, and
   ≤ 2500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-rack connections (including cable resistance), all inter-tier connections (including cable resistance), and all field terminal connections at the battery.

e. At least once per 18 months by verifying that the battery charger will supply at least 170 amperes at 125 volts for at least 4 hours.

f. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

g. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.3.2.f if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

h. At least once per 12 months, during shutdown, if the battery shows signs of degradation OR has reached 85% of the service life with a capacity less than 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

i. At least once per 24 months, during shutdown, if the battery has reached 85% of the service life with capacity greater than or equal to 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

3/4 8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.1 All containment penetration conductor overcurrent protective devices required to provide thermal protection of penetrations shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping either the primary or backup protective device, or racking out or removing the primary or backup device within 72 hours, declare the affected system or component inoperable, and verify the primary or backup protective device to be tripped, or the primary or backup device racked out or removed at least once per 7 days thereafter; or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 All required containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. For at least one 4.16 KV reactor coolant pump circuit, such that all reactor coolant pump circuits are demonstrated OPERABLE at least once per 72 months, by performance of:

      (a) A CHANNEL CALIBRATION of the associated protective relays, and

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

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. 278
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 14, 2009, as supplemented by letter dated April 12, 2010, 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. 278, 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: June 15, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 278

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following pages of Facility Operating License No. DPR-75 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
Page 4	Page 4
Page 5	Page 5-6
Page 6	- - -
Page 7	Page 7
Page 8-9-10	Page 8 THROUGH 20
Page 11	- - -
Page 12	- - -
Page 13	- - -
Page 14	- - -
Page 15	- - -
Page 16	- - -
Page 17	- - -
Page 18	- - -
Page 19	- - -
Page 20	- - -
Page 21	Page 21

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 2-1	3/4 2-1
3/4 2-7	3/4 2-7
3/4 3-39a	3/4 3-39a
3/4 3-66	3/4 3-66
3/4 4-33	3/4 4-33
6-21	6-21
6-24	6-24

(2) Technical Specifications and Environmental Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 278, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

PAGES 5 AND 6 ARE INTENTIONALLY BLANK
ITEMS 3 THROUGH 9 DELETED

(10) Fire Protection

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, as approved in the NRC Safety Evaluation Report, dated November 20, 1979, and in its supplements, and in the NRC Safety Evaluation dated January 7, 2004, 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.

PAGES 8 THROUGH 20 ARE INTENTIONALLY BLANK
ITEMS 11 THROUGH 25 DELETED

(26) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 227 are hereby incorporated into this license. PSEG Nuclear LLC shall operate the facility in accordance with the Additional Conditions.

(27) PSE&G TO PSEG Nuclear LLC License Transfer Conditions

a. PSEG Nuclear LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application, the requirements of the Order Approving Transfer of License and Conforming Amendment, dated February 16, 2000, and the related Safety Evaluation dated February 16, 2000.

b. The decommissioning trust agreement shall provide that:

1) The use of assets in both the qualified and non-qualified funds shall be limited to expenses related to decommissioning of the unit as defined by the NRC in its regulations and issuances, and as provided in the unit's license and any amendments thereto. However, upon completion of decommissioning, as defined above, the assets may be used for any purpose authorized by law.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE shall be maintained within the target band about the target flux difference as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference as specified in the COLR and with THERMAL POWER:

   1. Above 90% of RATED THERMAL POWER, within 15 minutes:

      a) Either restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

   2. Between 50% and 90% of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the target band as specified in the COLR for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits as specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

      b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits as specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

* See Special Test Exception 3.10.2

SURVEILLANCE REQUIREMENTS (Continued)

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional core power distribution measurements shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the COLR per specification 6.9.1.9.

f. The F_{xy} limits of e., above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0% to 15%, inclusive.

2. Upper core region from 85% to 100%, inclusive.

3. Grid plane regions at 17.8% ± 2%, 32.1% ± 2%, 46.4% ± 2%, 60.6% ± 2% and 74.9% ± 2%, inclusive.

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a core power distribution measurement and increased by the applicable manufacturing and measurement uncertainties as specified in the COLR.

TABLE 3.3-6 (Continued)
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	≤ 3.0×10^{-2} µCi/cm^3 (Alarm only)	10^{-3}-10^{1} µCi/cm^3	26
2) High Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	≤ 1.0×10^{2} µCi/cm^3 (Alarm only)	10^{-1}-10^{5} µCi/cm^3	26
3) Condenser Exhaust System	1	1,2,3&4	≤ 7.12×10^{4} cpm (Alarm only)	1-10^{6} cpm	26
3. CONTROL ROOM					
a. Air Intake - Radiation Level	2/Intake##	**	≤ 2.48×10^{3} cpm	10^{1}-10^{7} cpm	27,28

Control Room air intakes shared between Unit 1 and 2.

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILTY: MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.14.1 The operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, and 3.3.3.14.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.14.2 Calibration of the PDMS is required:

a. At least once every 180 Effective Full Power Days when the minimum number and core coverage criteria as defined in 3.3.3.14.b.1 and 3.3.3.14.b.2 are satisfied, or

b. At least once every 31 Effective Full Power Days when only the minimum number criterion as defined in 3.3.3.14.b.3 is satisfied.

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 Specification 4.0.5, 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.

6.9.1.5 Reports required on an annual basis shall include:

a. DELETED

b. DELETED

c. The results of any specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.9. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 DELETED

6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

   2. Control Bank Insertion Limits for Specification 3/4.1.3.5,

   3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,

   4. Heat Flux Hot Channel Factor, F_Q, its variation with core height, K(z), and Power Factor Multiplier PF_{xy}, Specification 3/4.2.2, and

   5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

   6. Refueling boron concentration per Specification 3.9.1

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   1. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (W Proprietary), Methodology for Specifications listed in 6.9.1.9.a.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 295 AND 278 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 14, 2009, as supplemented by letter dated April 12, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML092680244 and ML101100469, respectively), PSEG Nuclear, LLC (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 requested changes would make miscellaneous administrative and editorial changes to the TS and FOLs including correction of typographical and format errors, correction of administrative differences between units, and deletion of historical requirements that have expired.

The supplement dated April 12, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 17, 2009 (74 FR 59262).

2.0 REGULATORY EVALUATION

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

On July 22, 1993 (58 FR 39132), the Commission published a "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement) which discussed the criteria to determine which items are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953, July 19, 1995). Specifically, 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

> Criterion 1
> Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> Criterion 2
> A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 3
> A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 4
> A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

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

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

3.0 TECHNICAL EVALUATION

3.1 TS 3.1.3.3 Action b (Unit 1) - Delete TS 3.1.3.3 Action b

TS 3.1.3.3 Action b provides requirements if rod drop time is determined with 3 reactor coolant pumps running in Modes 1 and 2. However, TS 3.4.1.1 requires that all reactor coolant pumps (i.e., 4 pumps) be in operation in Modes 1 and 2. By previous amendments[1] 3-loop operation references were removed from the TSs. However, TS 3.1.3.3 Action b was not requested to be deleted in the associated applications for these amendments due to an oversight. The proposed amendment would delete TS 3.1.3.3 Action b.

---

1 Unit 1 Amendment No. 201 dated November 26, 1997 (ADAMS Accession No. ML011720441), and Unit 2 Amendment No. 197 dated January 8, 1999 (ADAMS Accession No. ML011730279).

The NRC staff concludes that since having 3 reactor coolant pumps running in Modes 1 and 2 is prohibited by TS 3.4.1.1 and that 3-loop operation references were previously removed from the TSs, the proposed change is administrative in nature, and therefore is acceptable.

The NRC staff notes that the application dated September 14, 2009, requested that the proposed change to TS 3.1.3.3 be made for both Salem Units 1 and 2. However, the proposed change for Unit 2 was withdrawn by the supplement dated April 12, 2010, to avoid an implementation conflict with another amendment which changed the same TS page.

3.2 LCO 3.2.1 (Unit 2) - Correct Typographical Error

The licensee proposes to correct a typographical error in which the "Core Operating Limits Report" is abbreviated as "CORL." The proposed change is to substitute "COLR" for "CORL." The NRC staff concludes that this change is administrative in nature, and therefore is acceptable.

3.3 SR 4.2.2.2 (Units 1 and 2) - Correct Subscript and Superscript Errors

The peaking factor term in SR 4.2.2.2 is represented in some cases with the superscript before the subscript. The licensee proposes to correct those cases and have the subscript before the superscript.

The NRC staff concludes that since this change would make the representations of the peaking factor term uniform, it is administrative and in nature, and therefore is acceptable.

3.4 SR 4.2.2.2 (Unit 2) - Delete Historical Cycle 11 Footnote

SR 4.2.2.2 contains a footnote with a requirement that pertained to a previous operating cycle (Cycle 11). The licensee proposed to remove this footnote.

The NRC staff concludes that this change is acceptable because the footnote does not contain any requirements applicable to current or future operations.

3.5 TS Table 3.2-1 (Unit 1) - Delete the Word "Increase" (2 instances) from the Footnote

A footnote for TS Table 3.2-1 (Unit 1) contains two instances of the word "increase" which are not present in the corresponding Unit 2 TS Table 3.2-1. The licensee proposes to delete those two instances since the word is superfluous and inconsistent with the corresponding footnote for Unit 2. The footnote currently reads as follows:

> Limit not applicable during either THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

As shown above, the footnote contains the phrase "in excess" following the word "increase" in both instances. As such, the NRC finds that the word "increase" is not needed to understand the meaning of the footnote. The NRC staff concludes that this change is administrative in nature, and therefore is acceptable.

3.6 Table 3.3-6 (Units 1 and 2) - Correct Format and Typographical Errors in Table 3.3-6

During preparation of a license amendment request dated May 1, 2006 (ADAMS Accession No. ML061300615), the licensee made errors in reproducing two of the marked up TS pages included in the application showing changes to TS Table 3.3-6 (TS page 3/4 3-36a for Unit 1 and TS page 3/4 3-39a for Unit 2). Specifically, the pages in the application dated May 1, 2006, did not match the current NRC-approved TS pages at that time issued in Amendment Nos. 225 (Unit 1) and 206 (Unit 2) dated November 2, 1999 (ADAMS Accession No. ML993240414). The errors in the May 1, 2006, application were subsequently included in the TS pages issued in Amendment Nos. 280 (Unit 1) and 263 (Unit 2) dated April 19, 2007 (ADAMS Accession No. ML070920306).

The errors caused certain symbols (≤, μ) to be missing or shown as other symbols. In addition, some text was misaligned (i.e., not in the proper columns) and some heading underlines were missing. The NRC staff concludes that these changes are administrative in nature, and therefore are acceptable.

3.7 Table 4.3-3 (Unit 1) - Correct Footer Alignment

On the left side of the footer for Table 4.3-3 (Unit 1) on page 3/4 3-38a, the number "1" is missing following the word "Unit" (the "1" had been incorrectly tabbed over to be next to the page number in the center of the footer). The licensee proposes to correct the footer alignment such that the "1" is moved next to the word "Unit."

The NRC staff concludes that this change is administrative in nature, and therefore is acceptable.

3.8 TS 3.3.3.14 (Units 1 and 2) - Remove Reference to Deleted TS 3.3.3.2

The Action statement for TS 3.3.3.14 (Units 1 and 2), "Power Distribution Monitoring System" (PDMS), currently reads, in part, as follows:

> With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2, to obtain any required core power distribution measurements.

Amendment Nos. 282 (Unit 1) and 265 (Unit 2) dated June 6, 2007 (ADAMS Accession No. ML071200374) approved the relocation TS 3/4.3.3.2, "Moveable Incore Detectors," to the Updated Final Safety Analysis Report. Due to an oversight, the licensee's application associated with Amendment Nos. 282 and 265 did not request removal of the reference to TS 3.3.3.2 in TS 3.3.3.14.

The proposed amendment would remove the reference to the deleted TS as follows:

> With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS

inoperable and use the incore movable detector system~~, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2,~~ to obtain any required core power distribution measurements.

The NRC staff concludes that since TS 3.3.3.2 has been removed, the change is administrative in nature, and therefore is acceptable.

3.9 SR 4.4.11.2 (Unit 2) - Delete Historical SR

SR 4.4.11.2 (Unit 2) requires that the No. 21 steam generator channel head be ultrasonically inspected "during each of the first three refueling outages." The licensee proposed to remove this SR since it is historical and is no longer applicable.

The NRC staff concludes that this change is acceptable because the SR does not contain any requirements applicable to current or future operations.

3.10 TS 6.9.1.5.c (Unit 2) - Correct Reference to TS 3.4.9

TS 6.9.1.5.c (Unit 2) requires, in part, that reports be submitted to the NRC on an annual basis specifying the "results of any specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8." The licensee's application states that TS 6.9.1.5.c should reference TS 3.4.9, "Reactor Coolant System - Specific Activity," not TS 3.4.8.

The Salem Unit 2 TSs currently does not have a TS 3.4.8. That TS was deleted on February 22, 1996, in Salem Unit 2 Amendment No. 161 (ADAMS Accession No. ML011720594). The NRC staff agrees with the licensee that TS 6.9.1.5.c should reference TS 3.4.9 which contains limits on the specific activity for the primary coolant. The NRC staff concludes that this change is administrative in nature, and therefore is acceptable.

3.11 TS 6.9.1.9.a.1 (Unit 2) - Correct Reference to TS 3/4.1.1.3

TS 6.9.1.9 (Unit 2) provides requirements regarding the content of the COLR. TS 6.9.1.9.a.1 requires that the following limits be documented in the COLR:

> Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.4,

The licensee's application states that TS 6.9.1.9.a.1 should reference TS 3/4.1.1.3, not TS 3/4.1.1.4. For Unit 2, the moderator temperature coefficient limits are controlled by TS 3/4.1.1.3. TS 3/4.1.1.4 applies to the minimum temperature for criticality. The NRC staff agrees that the correct reference is TS 3/4.1.1.3. The NRC staff concludes that this change is administrative in nature, and therefore is acceptable.

3.12 Various (Unit 1) - Delete Historical Cycle 13 Notes

Amendment No. 222 for Salem Unit 1 dated May 4, 1999 (ADAMS Accession No. ML011730078), allowed a one-time extension of the TS surveillance interval for certain SRs. The amendment added footnotes pertaining only to Salem Unit 1 Cycle 13 for the affected SRs.

The licensee proposed to delete the Cycle 13 footnotes since they are historical and are no longer applicable. The proposed amendment would revise the following TS pages: 3/4 1-9, 3/4 3-13, 3/4 3-31a, 3/4 3-57a, 3/4 5-2, 3/4 5-5, 3/4 7-21, 3/4 7-34, 3/4 8-9a, and 3/4 8-14.

The NRC staff concludes that this change is acceptable because the footnotes proposed for deletion do not contain any requirements applicable to current or future operations.

3.13 TS 3.1.3.2.1, 4.1.3.1.1 and 4.1.3.4 (Unit 1) - Delete Historical Cycle 14 Notes

Amendment 230 for Salem Unit 1 dated May 26, 2000 (ADAMS Accession No. ML003719424), added footnotes to TS 3.1.3.2.1, 4.1.3.1.1 and 4.1.3.4 associated with the methods and timeframes to determine the position of Rod 1SB2. These notes were put in place to facilitate repairs to the rod position indication system and were applicable only during Cycle 14. The licensee proposed to delete the Cycle 14 footnotes since they are historical and are no longer applicable.

The NRC staff concludes that this change is acceptable because the footnotes proposed for deletion do not contain any requirements applicable to current or future operations.

3.14 FOL Attachment 1, "Incomplete [P]reoperational Tests, Startup Tests, and Other Items Which Must be Completed" (Unit 1) - Delete Historical Attachment and Reference to It

Attachment 1 to the Salem Unit 1 FOL, "Incomplete [P]reoperational Tests, Startup Tests, and Other Items Which Must be Completed" consists of a listing of required actions related to the initial plant startup. The licensee proposed to delete the attachment and the reference to the attachment on page 5c of the FOL since the requirements in the attachment are historical and are no longer applicable.

The NRC staff concludes that this change is acceptable because Attachment 1 to the Salem Unit 1 FOL does not contain any requirements that are applicable to current or future operations.

3.15 FOL Conditions 2.C.3 through 2.C.9 and 2.C.11 through 2.C.25 (Unit 2) - Delete Historical Requirements

Salem Unit 2 FOL Conditions 2.C.3 through 2.C.9, 2.C.11, 2.C.12, and 2.C.14 through 2.C.25 all relate to actions required within specific time frames or dates. These time frames and dates have all now passed. The licensee proposed to delete these license conditions since they are historical and are no longer applicable.

The NRC staff concludes that deleting Salem Unit 2 FOL Conditions 2.C.3 through 2.C.9, 2.C.11, 2.C.12, and 2.C.14 through 2.C.25 is acceptable because they do not contain any requirements that are applicable to current or future operations.

The licensee also proposed to delete Salem Unit 2 FOL Condition 2.C.13 since it is historical and is no longer applicable. Condition 2.C.13 does not include a specific time frame or date. It states: "PSE&G shall also report for the Salem facility any information reported for the Hope Creek facility relating to circumstances which suggest that the risk from flammable gas clouds (resulting from river traffic accidents on the Delaware River) varies significantly from that previously considered."

The Hope Creek Generating Station (Hope Creek) is located adjacent to Salem Units 1 and 2. During construction of Hope Creek, issues were raised regarding the likelihood that a cloud of flammable gas might reach the site as a result of the accidental release of liquefied natural gas (LNG), or a similar highly flammable gas, following a postulated tanker accident on the Delaware River.

On February 1, 1979, the NRC staff issued an amendment to the construction permits for Hope Creek (ADAMS Accession No. ML011760592). The amendment added conditions to the permits designed to ensure that the NRC staff will be promptly alerted should circumstances arise which suggest that either LNG traffic or a significant increase in liquefied petroleum gas traffic on the Delaware River will materialize, or that other factors which govern the flammable vapor cloud probability calculation will change. These conditions were based on a decision by the Atomic Safety and Licensing Appeal Board (ASLAB) dated January 12, 1979 (ADAMS Legacy Library Accession No. 7901230367). The ASLAB decision indicated that, based on the low probability of a flammable vapor cloud reaching the site, construction of Hope Creek may continue without modification to the design. However, the ASLAB directed that this issue be reassessed by the applicant and the NRC staff at the operating license review stage.

On May 20, 1981, the NRC staff issued the operating license for Salem Unit 2. The operating license included Condition 2.C.13 consistent with the issues raised during construction of Hope Creek.

As discussed in Section 2.2.2 of NUREG-1048, "Safety Evaluation Report related to the operation of Hope Creek Generating Station," dated October 1984 (ADAMS Accession No. ML091310373), the applicant provided updated data indicating that the number of ships carrying liquefied flammable gases past the site was significantly less than the number of ships assumed in the analysis during the construction review stage. As such, the NRC staff concluded that the probability of waterborne transportation accidents causing radiological consequences in excess of the guidelines of 10 CFR Part 100 is within the acceptance criteria of Standard Review Plan Section 2.2.3. As such, no license conditions were included in the Hope Creek operating license requiring the reporting of circumstances which suggest that the risk from flammable gas clouds (resulting from river traffic accidents on the Delaware River) varies significantly from that previously considered.

The action required in Salem Unit 2 FOL Condition 2.C.13 (i.e., report to NRC regarding change in risk of flammable gas clouds) is dependent on the same information being reported for Hope Creek. However, Hope Creek is no longer is required to report this information. As such, the NRC staff concludes that the deletion of Condition 2.C.13 is acceptable since it is historical and is no longer applicable.

3.16 Technical Evaluation Conclusion

Based on the considerations in Sections 3.1 through 3.15 above, 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 surveillance requirements. 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 59262). 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.

Principal Contributors: L. Gibson
R. Ennis

Date: June 15, 2010

June 15, 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: MISCELLANEOUS ADMINISTRATIVE AND EDITORIAL CHANGES (TAC NOS. ME2229 AND ME2230)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment Nos. 295 and 278 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 14, 2009, as supplemented by letter dated April 12, 2010.

The amendments make miscellaneous administrative and editorial changes to the TS and FOLs including correction of typographical and format errors, correction of administrative differences between units, and deletion of historical requirements that have expired.

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. 295 to License No. DPR-70
2. Amendment No. 278 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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