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

August 16, 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 NO. 1, ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF THE TYPE A INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. ME2258)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 296 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 21, 2009, as supplemented by letter dated February 24, 2010.

The amendment revises TS 6.8.4.f, "Primary Containment Leakage Rate Testing Program," to allow a one-time extension of the Type A integrated leak rate test (ILRT) interval from 10 to 15 years. Specifically, the amendment requires that the next Type A ILRT be performed no later than May 7, 2016.

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 No. 50-272

Enclosures:
1. Amendment No. 296 to License No. DPR-70
2. 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. 296
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 21, 2009, as supplemented by letter dated February 24, 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. 296, 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 16, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 296

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 page of the Appendix A, Technical Specifications, 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
6-19	6-19

(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. 296, 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.

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring

e. Deleted

6.8.4.f. Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after May 7, 2001, shall be performed no later than May 7, 2016.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 296 TO FACILITY OPERATING LICENSE NO. DPR-70

PSEG NUCLEAR, LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-272

1.0 INTRODUCTION

By letter dated September 21, 2009, as supplemented by letter dated February 24, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML092730362 and ML100630695, respectively), PSEG Nuclear, LLC (PSEG or the licensee) submitted a request for changes to the Salem Nuclear Generating Station (Salem), Unit No. 1, Technical Specifications (TSs). The proposed amendment would revise TS 6.8.4.f, "Primary Containment Leakage Rate Testing Program," to allow a one-time extension of the Type A integrated leak rate test (ILRT) interval from 10 to 15 years. A Type A test is an overall (integrated) leakage rate test of the containment structure.

The last Type A ILRT was performed on May 7, 2001, and the current 10-year interval for completion of the next Type A ILRT ends on May 7, 2011. The proposed amendment would allow the next Type A ILRT to be performed no later than May 7, 2016. The licensee's application dated September 21, 2009, stated that the one-time extension would provide substantial benefits in the form of reduced personnel exposure and reduced outage costs.

The supplement dated February 24, 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

Paragraph 50.54(o) of Title 10 of the *Code of Federal Regulations* (10 CFR), and 10 CFR Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. The Type A test must be conducted: (1) after a containment system has been completed and is ready for operation; and (2) at a periodic interval based on historical performance of the overall containment system. Section V.B.3 of 10 CFR 50, Appendix J, Option B, requires that the regulatory guide (RG) or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in an RG.

Salem Unit 1 TS 6.8.4.f, "Containment Leakage Rate Testing Program," requires that leakage rate testing be performed as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

NEI 94-01 specifies an initial test interval of 48 months for Type A tests, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at Salem Unit 1 have been successful, so the current interval requirement would normally be 10 years. However, by the application dated September 21, 2009, the licensee is seeking a deviation from the NEI 94-01 guidelines by requesting a one-time extension of the Type A test interval from 10 to 15 years based on historical performance of its containment supported by a risk-informed analysis. Specifically, the licensee is requesting a change to TS 6.8.4.f, which would require that "The first Type A test performed after May 7, 2001, shall be performed no later than May 7, 2016."

The 10 CFR Part 50, Appendix J, Option B local leakage rate tests (LLRTs) (Type B and Type C tests), including their schedules, are not affected by this amendment request.

The requirements in 10 CFR 50.55a address the use of codes and standards as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. Specifically, 10 CFR 50.55a(b)(2)(viii) identifies the regulatory conditions that apply to the use of Subsection IWL of Section XI of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) for the examination of concrete containment structures. Additionally, 10 CFR 50.55a(b)(2)(ix) identifies the regulatory conditions that apply to the use of Subsection IWE of Section XI of the ASME Code for the examination of metal containments and the liners of concrete containments.

3.0 TECHNICAL EVALUATION

3.1 Background

The Salem Unit 1 containment is a reinforced concrete structure with a welded carbon steel liner on the inside surface. The containment consists of a cylindrical wall, a flat base and a hemispherical dome, and penetrations through the structure. The steel liner and its penetrations establish the leakage limiting boundary of the containment. The leak-tight integrity of the containment penetrations (equipment hatch, airlocks, flanges, and sealing mechanisms) and isolation valves are verified through Type B and Type C LLRTs. The overall leak-tight integrity and structural integrity of the containment is verified through a Type A ILRT. These tests are performed at the calculated peak containment internal pressure related to the design-basis loss-of-coolant accident (LOCA). This pressure is referred to as P_a. For Salem Unit 1, P_a is

47.0 pounds per square inch (psig) as shown in TS 6.8.4.f. The maximum allowable containment leakage rate, when tested at P_a, is referred to as L_a. In accordance with TS 6.8.4.f, L_a is 0.1% of primary containment air weight per day.

The leakage rate testing requirements of 10 CFR Part 50, Appendix J, Option B (Type A ILRT and Type B and Type C LLRTs) and the containment inservice inspection (ISI) requirements mandated by 10 CFR 50.55a, complement each other in ensuring the leak-tightness and structural integrity of the containment during its service life.

As discussed above, Salem Unit 1 currently has a Type A ILRT interval of 10 years. The licensee has requested a one-time 5 year extension of the Type A test interval from 10 to 15 years. The licensee justified the proposed change primarily based on: (1) the containment leakage testing program; (2) the containment ISI and coating inspection programs; and (3) a risk-informed analysis. The NRC staff reviewed the licensee's justifications in these three areas as discussed in Safety Evaluation (SE) Sections 3.2, 3.3, and 3.4 respectively.

3.2 Containment Leakage Testing Program

Type A Tests

The licensee's application dated September 21, 2009, provided results of previous Type A ILRTs performed at Salem Unit 1. Prior to first plant operation, the licensee performed a one-time test by pressurizing the containment with air for a minimum period of 1-hour at 54 psig (115% of the design pressure) to verify structural integrity. In addition, three Type A ILRTs were performed in accordance with the TSs (in December 1987, April 1991 and May 2001). All tests passed the as-found acceptance criteria of 1.0 L_a. Therefore, the NRC staff concludes that the test results demonstrate acceptable performance of the Salem Unit 1 containment structure historically with respect to leakage integrity.

Type B and C Tests

The licensee's application dated September 21, 2009, provided a history of Type B and C tests of the last six Salem Unit 1 outages which includes the maximum penetration pathway as-left leakage. In all outages the maximum penetration pathway leakage was significantly less than the TS allowable value of 0.6 L_a.

The licensee also provided a comprehensive table that identified all the penetrations subjected to Type B and C testing and their current test frequencies. The licensee stated that the test frequencies are established based on performance utilizing the requirements of 10 CFR Part 50, Appendix J, Option B. The licensee indicated that the test frequencies are re-evaluated after each refueling outage for potential changes. The licensee also provided the date of the last test and the due date for the next scheduled test between now and the next ILRT. The licensee also noted that some of the scheduled test dates will be modified to ensure that penetrations and components not drained and vented during the next scheduled ILRT have current test results within the previous 24-month period to meet the requirements of NEI 94-01, Section 9.2.1.

Based on review of the licensee's test schedule information, the NRC staff finds that the leakage performance of each of the containment pressure boundary penetrations will be adequately monitored by a Type B or Type C test during the requested extension period for the ILRT interval. In addition, based on the past test results and the planned testing, the NRC staff finds that the licensee is effectively implementing its Type B and Type C LLRT program, under 10 CFR Part 50, Appendix J, Option B, in a rational and systematic manner that is consistent with industry standards and regulatory requirements, and there is reasonable assurance it will continue to do so during the requested ILRT interval extension period.

Visual Examinations

Option B of Appendix J to 10 CFR Part 50 requires that a general visual inspection of the accessible interior and exterior surfaces of the containment system be conducted, prior to each Type A test and at a periodic interval between tests based on performance of the containment system, for identification of structural deterioration which may affect the containment leak-tight integrity. RG 1.163, Regulatory Position C.3, states that these examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval has been extended to 10 years.

As discussed in the licensee's application dated September 21, 2009, the licensee stated that the visual examination conducted during the spring 2007 refueling outage was completed satisfactorily. The licensee also provided the schedule for visual examination during the proposed extended interval. The schedule includes three visual examinations planned during the extended interval. In a request for additional information (RAI), the licensee was requested to discuss if the station procedure SH.RA-ST.ZZ-0106 used for the spring 2007 examination and also to be used for future examinations is in accordance with the requirements of Appendix J to 10 CFR Part 50 and the guidance in RG 1.163 regarding general and visual examination. In the supplement dated February 24, 2010, the licensee confirmed that this procedure is in accordance with the requirements in Appendix J to 10 CFR Part 50 and the guidance in RG 1.163 regarding general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity. The procedure identifies the specific interior and exterior areas to be inspected, and the notifications to be generated as a result of these inspections that require further review or evaluation. The procedure allows credit to be taken for the ASME Code, Section XI, IWE/IWL exams performed during the same refueling outage.

The NRC staff finds that the licensee's program for visual examination of the containment provides reasonable assurance that structural deterioration which may affect the containment leak-tight integrity will be detected consistent with the intent of the requirements of Appendix J to 10 CFR Part 50 and the guidance in RG 1.163.

3.3 Containment ISI and Coating Inspection Programs

The Salem Unit 1 containment inservice inspection (CISI) program provides requirements for examination of the metal containment liner (in accordance with Subsection IWE of Section XI of the ASME Code) and the containment concrete (in accordance with Subsection IWL of Section XI of the ASME Code). The licensee's application dated September 21, 2009, provided a discussion regarding the CISI program requirements and history as described below.

IWE Examinations (Metal Containment Liner)

For the first CISI interval, which ended in April 2010, the IWE examinations at Salem Unit 1 were performed in accordance with the 1998 Edition, 1998 Addenda of the ASME Code Section XI, as modified by 10 CFR 50.55a and Relief Request No. RR-E1, which was authorized by the NRC staff on June 6, 2000 (ADAMS Accession No. ML003720636). For the current (second) CISI interval, which includes the requested 5-year extension period, the examinations will be performed in accordance with the 2004 Edition of the ASME Code, Section XI.

The licensee's application dated September 21, 2009, provided the following summary of the results and corrective actions of the IWE examinations that have been performed:

> The first IWE examination was performed in spring 2001 (1R14) and resulted in no reportable indications. Several Notifications were processed to document examination indications of coating degradation on containment penetrations and areas on the metal containment liner during this examination campaign. Engineering evaluations were performed on noted areas of degradation and all areas were found acceptable. Corrective maintenance orders were generated to restore the degraded coatings to original configuration. A number of broken or missing liner insulation retaining studs were also identified. An evaluation determined that the missing studs did not adversely affect the structural integrity of the containment liner and had no affect on moisture intrusion for the liner.
>
> The second IWE examination was performed in spring 2004 (1R16) and resulted in no reportable indications. Several Notifications were processed to document examination indications of coating degradation and blistering on containment penetrations and on the metal containment liner during this examination campaign. Engineering evaluations were performed on noted areas of degradation and all areas were found acceptable. Corrective maintenance orders were generated to restore the degraded coatings to original configuration.
>
> The third and most recent IWE examination was performed in spring 2007 (1R18). No reportable or recordable indications were identified.

The next scheduled IWE examinations are in fall 2011 and fall 2014. The last IWE examination during the second CISI interval is planned for spring 2019, which is beyond the requested extension period.

In an RAI, the licensee was requested to describe the engineering evaluations and the acceptance criteria for the containment penetrations and metal liner coating degradation noted during the first and second IWE examinations. In the supplement dated Febraury 24, 2010, the licensee stated that the engineering evaluations and acceptance criteria were in accordance with the requirements of IWE-3122.3 of the 1998 Addenda of the ASME Code, Section XI. The evaluations addressed the structural integrity of the containment, including the condition of the base metal beneath the degraded coating areas. A detailed visual examination of the base metal in these local areas was performed, with the conclusion that any material loss was less than 10% of the nominal plate thickness criterion, supporting the conclusion that the components were acceptable by engineering evaluation.

IWL Examinations (Containment Concrete)

For the first CISI interval, which ended in April 2010, the IWL examinations at Salem Unit 1 were performed in accordance with the 1998 Edition, 1998 Addenda of the ASME Code Section XI, as modified by 10 CFR 50.55a and Relief Request No. RR-L1, which was authorized by the NRC staff on June 6, 2000 (ADAMS Accession No. ML003720636). For the current (second) CISI interval, which includes the requested 5-year extension period, the examinations will be performed in accordance with the 2004 Edition of the ASME Code, Section XI.

The licensee's application dated September 21, 2009, provided the following summary of the results and corrective actions of the IWL examinations that have been performed:

> The first IWL examination was performed in spring 2001 (1R14) resulting in no reportable or recordable indications.
>
> The second and most recent IWL examination performed during 1 R17 (Fall 2005) resulted in no reportable indications being identified. Examination revealed some acceptable minor surface scaling and identified moisture/intrusion barrier plate bolt coating having light to medium rust. Restoration of bolt coating is planned for 1 R20 (Spring 2010).
>
> To support license renewal activities, during 1R19 (Fall 2008), insulation panels were removed to permit the liner to be inspected in four normally inaccessible areas. No rejectable areas were identified. Light rusting was observed in two of the areas, but did not require any further action. All four locations examined were considered satisfactory for continued operation.

The next scheduled IWL examinations are in fall 2014 and spring 2019.

Inaccessible Areas

In response to an NRC RAI, the licensee, in its supplement dated February 24, 2010, stated that during implementation of the first 10-year interval of the IWE/IWL CISI program, there were no instances where the existence of, or potential for, degraded conditions in inaccessible areas of the containment structure and metallic liner were identified or evaluated based on conditions found in accessible areas as required by 10 CFR 50.55a (b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A).

The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. This issue is discussed below in SE Section 3.4.

Coating Inspection Program

As discussed in the application dated September 21, 2009, PSEG has implemented controls for the procurement, application, and maintenance of Service Level I protective coatings used in containment, consistent with the licensing basis and regulatory requirements applicable to the Salem Station. As defined in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Revision 1, dated July 2000, Service Level I coatings are used in areas inside the reactor containment where the coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown. PSEG's containment coatings monitoring program is based on the guidance of American Society for Testing and Materials (ASTM) D5163 and is consistent with the guidance in RG 1.54. Defects observed during periodic visual examinations are documented in the PSEG corrective action program, assessed, and repaired or replaced as necessary.

Conclusion

Based on the discussion above, the NRC staff finds that the licensee has implemented programs that adequately examine, monitor and manage degradation of the Salem Unit 1 containment structure. Furthermore, the staff finds that the Salem Unit 1 CISI and coating inspection programs complement the containment leakage testing program (discussed above in SE Section 3.2) in assuring the structural integrity and leak-tightness of the containment.

3.4 Risk-Informed Analysis

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the application dated September 21, 2009. In performing the risk assessment, the licensee considered: (1) the guidelines of NEI 94-01, Revision 0; (2) the methodology used in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994; (3) the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Surveillance Intervals," dated November 13, 2001; (4) the risk assessment template in EPRI TR-1009325, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated October 2008; (5) the methodology used for Calvert Cliffs to assess the risk from undetected leaks due to corrosion; (6) RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002; and RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated January 2007. The licensee also performed an alternate quantification using the EPRI expert elicitation methodology. The NRC staff's evaluation and conclusion do not rely on this alternate assessment.

The basis for a 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J of 10 CFR Part 50. Section 11.0 of NEI 94-01 states that NUREG-1493, "A Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for Salem Unit 1 early in the plant's life required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, that was detectable only by a Type A test, goes undetected from 18 months to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the pressurized water reactor and boiling water reactor representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in increased public dose.

The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking was completed in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using the RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 states that a PRA used in risk-informed regulation should be performed in a manner that is consistent with accepted practices. In Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007, the NRC clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 to determine whether the technical adequacy of the PRA used to support a submittal is consistent with accepted practices. Revision 2 of RG 1.200 will be used for all risk-informed applications received after March 2010. In the Final Safety Evaluation for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, dated June 25, 2008, the NRC staff states that Capability Category I of the ASME PRA Standard shall be applied as the standard for assessing PRA quality for ILRT extension applications since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their contribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

PSEG's application dated September 21, 2009, and the supplement dated February 24, 2010, addressed the technical adequacy of the PRA which forms the basis for the subject risk assessment. As described therein, the current Salem Unit 1 PRA model used for the application is revision 4.2, completed March 2009. In 2009, a formal peer review team published its review (conducted in 2008) of the Salem Generating Station (SGS) PRA, using the PRA Standard, ASME-RA-Sb-2005, and RG 1.200, Revision 1. The license amendment request provided a summary of the eight "key findings" from this peer review, and an assessment of the impact of the key findings on the ILRT extension application. In response to an NRC staff RAI, the licensee also provided a list of all the supporting requirements that the 2008 peer review identified as "Not Met" and an assessment of the impact of these findings on the ILRT extension application, and additional information on several of the "key findings." The licensee's assessments concluded that the 2008 peer review findings have to do with documentation only, or the findings have no impact or no significant impact on the ILRT application. The NRC staff reviewed this information and agrees with the licensee's assessments. Given that a formal peer review team has evaluated the SGS PRA against RG 1.200 and the ASME PRA Standard, and given that the licensee evaluated all of the peer review findings for applicability to the ILRT extension and determined that any unresolved issues would not impact the conclusions of the

ILRT risk assessment, the NRC staff concludes that the Salem Unit 1 PRA model revision 4.2 is of sufficient technical quality to support the evaluation of changes to ILRT frequencies.

RG 1.174 provides risk-acceptance guidelines for assessing the increases in CDF and LERF for risk-informed license amendment requests. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also discusses defense-in-depth. The licensee estimated the change in the conditional containment failure probability for the proposed change and judged it to be insignificant and reflecting sufficient defense-in-depth.

The licensee comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The following is a summary of results from the licensee analysis associated with extending the Type A test frequency:

Total Integrated Plant Risk

Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be less than 1 person-rem per year, and about 1 percent of the total population dose. This increase is small, as defined in the NRC's June 25, 2008, final safety evaluation for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2.

Increase in LERF

The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be about 4.2×10^{-7} per year based on the plant-specific internal events PRA, and about 8.4×10^{-7} per year when external events are included. There is some likelihood that the flaws in the containment estimated as part of the Class 3b (large isolation failure - liner breach) frequency would be detected as part of the ASME Code, Section XI, Subsection IWE and IWL examinations of the containment surfaces. Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change, and the calculated increases in LERF reported above includes the impact of age-related corrosion. As discussed in the supplement dated February 24, 2010, the licensee considered instances of liner corrosion found over the past 10 years, and found that the sensitivity calculation in the application (resulting in an increase in LERF of about 3.6×10^{-8} per year) bounds the effect of including all known corrosion events.

Pursuant to RG 1.174, when the calculated increase in LERF is in the range of 10^{-7} per year to 10^{-6} per year, applications are considered if the total LERF is less than 10^{-5} per year. Based on information provided by the licensee, the total LERF for internal and external events, including the requested change, is estimated to be about 8.2×10^{-6} per year, which meets the total LERF criteria. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

Defense-in-Depth

RG 1.174 also provides guidance for the licensee to show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of approximately one percentage point for the cumulative change of going from a test frequency of three in 10 years to one in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

3.5 Technical Evaluation Conclusion

Based on the findings in SE Sections 3.2, 3.3, and 3.4, the NRC staff concludes that there is reasonable assurance that the containment structural and leak-tight integrity will continue to be maintained if the ILRT interval is extended on a one-time basis to 15 years. Therefore, the staff further 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 amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (74 FR 59262). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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

Principal Contributors: A. Sallman
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R. Ennis

Date: August 16, 2010

August 16, 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 NO. 1, ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF THE TYPE A INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. ME2258)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 296 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 21, 2009, as supplemented by letter dated February 24, 2010.

The amendment revises TS 6.8.4.f, "Primary Containment Leakage Rate Testing Program," to allow a one-time extension of the Type A integrated leak rate test (ILRT) interval from 10 to 15 years. Specifically, the amendment requires that the next Type A ILRT be performed no later than May 7, 2016.

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 No. 50-272
Enclosures:
1. Amendment No. 296 to License No. DPR-70
2. Safety Evaluation

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