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

September 8, 2015

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS RE: EXTENSION OF TYPE A AND TYPE C LEAK RATE TEST FREQUENCIES (TAC NOS. MF5172 AND MF5173)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment Nos. 302 and 306 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3. These amendments consist of changes to the Technical Specifications (TSs) and Facility Operating Licenses in response to your application dated November 7, 2014, as supplemented by letters dated April 13, 2015, and August 10, 2015.

The amendments revise the TSs associated with the primary containment leakage rate testing program. Specifically, the amendments extend the frequencies for performance of the Type A containment integrated leakage rate test and the Type C containment isolation valve leakage rate test, which are required by Title 10 of the *Code of Federal Regulations* Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

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

Sincerely,

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

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 302 to Renewed DPR-44
2. Amendment No. 306 to Renewed DPR-56
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 302
Renewed License No. DPR-44

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), and PSEG Nuclear LLC, dated November 7, 2014, as supplemented by letters dated April 13, 2015, and August 10, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 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;

   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.

Enclosure 1

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 302, are hereby incorporated in the license. Exelon Generation Company 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: September 8, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 302

RENEWED FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following page of the Renewed Facility Operating License with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

Replace the following page of the Appendix A Technical Specifications with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
5.0-17	5.0-17

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility, and such Class B and Class C low-level radioactive waste as may be produced by the operation of Limerick Generating Station, Units 1 and 2.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the Peach Bottom Atomic Power Station, Unit 2, at steady state reactor core power levels not in excess of 3951 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 302, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Peach Bottom Atomic Power Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 281 and modified by Amendment No. 301.

(4) Fire Protection

The Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report (SER) dated May 23, 1979, and Supplements dated August 14, September 15, October 10 and November 24, 1980, and in the NRC SERs dated September 16, 1993, and August 24, 1994, subject to the following provision:

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[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exception:

a. Section 10.2: MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.7% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and Type C tests and ≤ 0.75 L_a for Type A tests;

(continued)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 306
Renewed License No. DPR-56

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), and PSEG Nuclear LLC, dated November 7, 2014, as supplemented by letters dated April 13, 2015, and August 10, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 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;

   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.

Enclosure 2

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Renewed Facility Operating License No. DPR-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 306, are hereby incorporated in the license. Exelon Generation Company 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: September 8, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 306

RENEWED FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following page of the Renewed Facility Operating License with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

Replace the following page of the Appendix A Technical Specifications with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
5.0-17	5.0-17

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility, and such Class B and Class C low-level radioactive waste as may be produced by the operation of Limerick Generating Station, Units 1 and 2.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the Peach Bottom Atomic Power Station, Unit No. 3, at steady state reactor core power levels not in excess of 3951 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 306, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Peach Bottom Atomic Power Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 283 and modified by Amendment No. 304.

[1]-The Training and Qualification Plan and Safeguards Contingency Plan and Appendices to the Security Plan.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exception:

a. Section 10.2: MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.7% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for Type A tests;

(continued)

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 302 AND 306

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-44 AND DPR-56

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By application dated November 7, 2014 (Reference 1), as supplemented by letters dated April 13, 2015 (Reference 2), and August 10, 2015 (Reference 16), Exelon Generation Company, LLC (Exelon, the licensee), requested changes to the Technical Specifications (TSs) for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed amendments would revise the TSs associated with the primary containment leakage rate testing program. Specifically, the amendments would extend the frequencies for performance of the Type A containment integrated leakage rate test (ILRT) and the Type C containment isolation valve leakage rate test, which are required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

The supplements dated April 13, 2015, and August 10, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on January 20, 2015 (80 FR 2749).

2.0 REGULATORY EVALUATION

The licensee requested a change to the Renewed Facility Operating Licenses for PBAPS, Units 2 and 3, in accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit." The regulations in 10 CFR 50.54(o) require that the primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J to 10 CFR Part 50 includes two options: Option A - Prescriptive Requirements, and Option B - Performance-Based Requirements, either of which can be chosen for meeting the requirements of Appendix J.

The testing requirements in Appendix J ensure that leakage through the primary containment and related systems and components penetrating primary containment does not exceed allowable leakage rate value specified in the TSs or associated bases, and that integrity of the containment structure is maintained during its service life.

The licensee has adopted and has been implementing Option B for meeting the requirements of Appendix J. Option B of Appendix J specifies the performance-based requirements and criteria for preoperational and subsequent leakage-rate testing. These requirements are met by (1) performance of Type A tests to measure the containment system overall integrated leakage rate; (2) Type B pneumatic tests to detect and measure local leakage rates across pressure retaining leakage-limiting boundaries such as penetrations; and (3) Type C pneumatic tests to measure containment isolation valve leakage rates. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each boundary and isolation valve (for Type B and C tests), to ensure the integrity of the overall containment system as a barrier to fission product release. The leakage rate test results must not exceed the allowable leakage rate with margin as specified in the TSs. Option B also requires that a general visual inspection for structural deterioration of the accessible interior and exterior surfaces of the containment, which may affect the containment leak-tight integrity, be conducted prior to each Type A test, and at a periodic interval between tests, based on the performance of the containment system.

Section V.B.3 of 10 CFR Part 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 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.

The implementation document that is currently referenced in the PBAPS, TS 5.5.12, "Primary Containment Leakage Rate Testing Program," is RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 3). RG 1.163 endorsed Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995, as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B of 10 CFR Part 50, Appendix J, subject to four regulatory positions delineated in Section C of the RG. NEI 94-01, Revision 0, includes provisions that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

The requested change would revise TS 5.5.12, "Primary Containment Leakage Rate Testing Program," to require compliance with NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 6), in lieu of RG 1.163, including listed exceptions. Additionally, the change would require compliance with the limitations and conditions specified in Section 4.0 of the safety evaluation (SE) for NEI 94-01, Revision 2-A (Reference 4).

NEI 94-01, Revisions 2 and 3, have been reviewed by the NRC and approved for use. The final SE for Revision 2, issued by letter dated June 25, 2008 (Reference 5), documents the NRC's evaluation and acceptance of Revision 2, subject to six specific limitations and conditions listed in Section 4.1 of the SE for Type A tests. The final SE for Revision 3, issued by letter dated June 8, 2012 (Reference 8), includes two specific limitations and conditions listed in Section 4.0 of the SE for Type C tests. The approved versions of NEI 94-01, Revisions 2 and 3, incorporating the NRC staff SEs were issued as NEI 94-01 Revision 2-A (Reference 4) and NEI 94-01, Revision 3-A (Reference 6), respectively. Consistent with the requested change, the licensee's submittal was reviewed against the limitations and conditions presented in the SEs included in NEI 94-01, Revision 2-A and 3-A.

In accordance with the guidance in NEI 94-01, Revision 2-A, the licensee proposes to extend the containment Type A test interval from the current approved 10 years to 15 years, based on acceptable performance. This would allow the next Type A test to be performed within 15 years from the last test, instead of the current 10-year interval. The previous Type A tests were performed in December 2014 (PBAPS, Unit 2) and in October 2005 (PBAPS, Unit 3). The approval of the amendment would allow the next Unit 3 test to be performed no later than October 2020 instead of no later than October 2015, based on the current TS requirements.

In accordance with the guidance in NEI 94-01, Revision 3-A, the licensee proposes to extend the containment Type C test interval from the current approved 60 months to 75 months, with a permissible extension period of 9 months (total of 84 months) for non-routine emergent conditions, based on acceptable performance. This would allow the next Type C test to be performed within 75 months from the last test, instead of the current 60-month interval.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Changes

The licensee's proposal would revise the PBAPS leakage rate testing program by implementing the guidance in NEI 94-01, Revision 3-A and the conditions and limitations specified in NEI 94-01, Revision 2-A.

PBAPS, Units 2 and 3, TS 5.5.12, "Containment Leakage Rate Testing Program," currently states, in part:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, 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 exceptions to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J:
>
> a. Section 10.2: MSIV [main steam isolation valve] leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

For PBAPS, Unit 2, TS 5.5.12 includes the following additional exception:

b. Section 9.2.3: The first Type A test performed after the October 2000 Type A test shall be performed no later than October 2015.

For PBAPS, Unit 3, TS 5.5.12 includes the following additional exception:

b. Section 9.2.3: The first Type A test performed after the December, 1991 Type A test shall be performed no later than December, 2006.

In its letter dated November 7, 2014, the licensee proposed to delete PBAPS, Units 2 and 3, TS 5.5.12 exception (b) and replace the reference to RG 1.163 with a reference to NEI 94-01, Revisions 2-A and 3-A. The proposed change would revise TS 5.5.12 for both units to state, in part:

> A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exception:
>
> a. Section 10.2: MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

The license amendment application follows NEI 94-01, Revision 3-A, and the limitations and conditions of Section 4.0 of the NEI 94-01, Revision 2-A SE, and Section 4.0 of the NEI 94-01, Revision 3-A SE. The licensee proposes an extension of the Type A test interval, which is currently required by TSs to be performed at 10-year intervals of no longer than 15 years from the last Type A test (October 2000 and October 2005 for PBAPS, Units 2 and 3, respectively). To extend the Type A test interval, NEI 94-01, Revision 3-A, provides a guideline that the extension shall be based on two consecutive successful Type A tests (i.e., performance history) and other requirements stated in Section 9.2.3 of NEI 94-01, Revisions 2-A and 3-A. The NRC staff's review of the PBAPS Type A test performance history, with respect to meeting the Section 9.2.3 requirements and SE limitations and conditions, is presented in SE Section 3.2.1 below.

The licensee also proposes an extension of the Type C test interval. For PBAPS, Units 2 and 3, Type C tests are currently required to be performed at no longer than a 60-month interval. The proposed amendment would extend the Type C test interval to no longer than 75-months from the last Type C test. The NEI 94-01, Revision 3-A guidelines explain that extensions of Type C test intervals are allowed, based upon completion of two consecutive periodic as-found tests, where the results of each test are within a licensee's allowable administrative limits and other requirements stated in Section 10.2.3 in NEI 94-01, Revision 3-A. The NRC staff's review of the PBAPS Type C test performance history, with respect to meeting the Section 10.2.3 requirements and SE limitations and conditions, is presented in SE Section 3.2.2 below.

3.2 Deterministic Considerations: Structural and Leak Integrity of the Containment

3.2.1 Historical Type A Test Results

The maximum allowable primary containment leakage rate, L_a, currently specified in PBAPS TS 5.5.12, is 0.7 percent of containment air weight per day at the peak calculated containment internal pressure for a design-basis loss-of-coolant accident, P_a. By amendments issued on September 5, 2008, the PBAPS design-basis containment leak rate, L_a, was changed from a value of 0.5 percent to 0.7 percent of containment air weight per day at the peak calculated containment internal pressure for a design-basis loss-of-coolant accident.

In Section 3.2.4 of Attachment 1 to the LAR, the licensee presented the results of the historical Type A tests that are summarized in Tables 1 and 2 below.

Table 1: PBAPS, Unit 2, Type A ILRT History

Test Date	Leakage Rate(1) (Containment Air Weight (wt) %/day)
May 1973	0.127
June 1976	0.016
July 1980	0.105
June 1985	0.70
	Retest 0.0156(2)
January 1989	0.233
March 1991	0.2135
October 2000	0.3365

(1) On September 5, 2008, the PBAPS design-basis containment leakage rate, L_a, was changed from a value of 0.5 wt%/ day at containment peak pressure, to a value of 0.7 wt%/ day as currently shown in TS 5.5.12.

(2) The first attempted test was terminated primarily due to leakage through the valve packing of a torus vacuum relief valve, AO-2502B. Identified leakage sources were repaired and a second ILRT was completed successfully.

Table 2: PBAPS, Unit 3, Type A ILRT History

Test Date	Leakage Rate(1) (Containment Air Weight %/day)
February 1974	0.116
April 1977	1.129(2)
	Retest 0.322
September 1981	0.389(3)
	Retest 0.185
August 1983	0.784(4)
	Retest 0.105
January 1986	0.088

Test Date	Leakage Rate(1) (Containment Air Weight %/day)
November 1989	0.229
December 1991	0.139
October 2005	0.2781

(1) On September 5, 2008, the PBAPS design-basis containment leakage rate, L_a, was changed from a value of 0.5 wt%/ day at containment peak pressure, to a value of 0.7 wt%/ day as currently shown in TS 5.5.12.

(2) Analysis of the ILRT data indicated that leakage from the containment was approximately 10 standard cubic feet per minute. The leak was identified on the airside of a Torus water level instrument. The leak was isolated via the instrument root valve and the ILRT was completed successfully.

(3) The major source of leakage was identified as a missing O-ring on Pressure Transmitter PT-3-05-012C (Drywell Pressure Transmitter). Failure to install the O-ring was an activity-based omission during instrument maintenance. Isolation of the instrument resulted in leakage from this source to be approximately 25,000 standard cubic centimeters per minute (sccm). Following installation of the missing O-ring, the ILRT was completed successfully.

(4) The major source of leakage was identified as packing leakage from MO-3-10-034A (Residual Heat Removal (RHR) Loop A Full Flow Test Line Block Valve). The valve was repacked on backseat and the ILRT was completed successfully.

The results of the last two Type A ILRTs for PBAPS, Units 2 and 3, are less than the previous maximum allowable containment leakage rate of 0.5 percent of containment air weight per day and the current maximum allowable containment leakage rate of 0.7 percent of containment air weight per day at the test pressure of 49.1 pounds per square inch gauge (psig). As a result, since both tests for both units were successful, both units have been placed on extended ILRT frequencies of once per 10 years[1].

Based on the above, the staff concludes that since the last two Type A ILRTs for PBAPS, Units 2 and 3, were less than the design-basis leak rate, the guidelines in NEI 94-01, Revisions 2-A and 3-A, regarding acceptable performance history, has been met. In addition, the NRC staff concludes that the results of the Type A ILRTs provide reasonable assurance that containment overall leakage will be maintained below the design-basis leak rate consistent with the requirements in TS 5.5.12.

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[1] PBAPS, Unit 2, Amendment No. 276, issued on July 20, 2010, revised TS 5.5.12 to allow a one-time change in the ILRT interval from 10 years to 15 years. This amendment required that the licensee perform the first Type A test after the October 2000 test, no later than October 2015. The test was performed in December 2014.

3.2.2 Historical Type B and Type C Test Results

In Section 3.4.4 of Attachment 1 to the LAR, the licensee presented the results of its Type B and Type C testing. Tables 3 and 4 below provide local leak rate test (LLRT) data trend summaries for PBAPS since the performance of the Unit 2 LLRT in 2000 and the Unit 3 LLRT in 2005.

Table 3: PBAPS, Unit 2, Type B and Type C LLRT Combined As-Found (AF)/ As-Left (AL) Trend Summary

RFO	2000	2002	2004	2006	2008	2010	2012
AF Min Path (sccm)	25178	21791	21392	20240	27965	19761	25867
Fraction of L_a	0.1439	0.1245	0.1222	0.1156	0.1598	0.1192	0.1478
AL Max Path (sccm)	61171	59042	43644	53701	59033	63432	65937
Fraction of L_a	0.3495	0.3373	0.2493	0.3069	0.3373	0.3624	0.3768
AL Min Path (sccm)	20244	21379	18937	28108	20281	26689	29089
Fraction of L_a	0.1157	0.1222	0.1082	0.1606	0.1159	0.1525	0.1662

Table 4: PBAPS, Unit 3, Type B and Type C LLRT Combined As-Found/ As-Left Trend Summary

RFO	2005	2007	2009	2011	2013
AF Min Path (sccm)	26436	17830	22495	18764	28513
Fraction of L_a	0.1511	0.1019	0.1285	0.1072	0.1629
AL Max Path (sccm)	47110	59954	46341	53562	54379
Fraction of L_a	0.2692	0.3426	0.2648	0.3061	0.3107
AL Min Path (sccm)	18221	23930	18962	26283	20070
Fraction of L_a	0.1041	0.1367	0.1083	0.1502	0.1147

A review of the Type B and Type C test results from 2000 through the fall of 2012 for PBAPS, Unit 2, and from 2005 through the fall of 2013 for PBAPS, Unit 3, has shown significant margin between the As-Found (AF) and As-Left (AL) outage summations and the TS limit of $\leq 0.6\ L_a$ as described below:

- The AF minimum pathway leak rate for PBAPS, Unit 2, shows an average of 22.1 percent of 0.6 L_a with a high of 26.6 percent of 0.6 L_a or 0.1598 L_a.

- The AL maximum pathway leak rate for PBAPS, Unit 2, shows an average of 55.2 percent 0.6 L_a with a high of 62.8 percent of 0.6 L_a or 0.3768 L_a.

- The AF minimum pathway leak rate for PBAPS, Unit 3, shows an average of 21.7 percent of 0.6 L_a with a high of 27.1 percent of 0.6 L_a or 0.1629 L_a.

- The AL maximum pathway leak rate for PBAPS, Unit 3, shows an average of 49.8 percent of 0.6 L_a with a high of 51.8 percent of 0.6 L_a or 0.3107 L_a.

The summary above shows that there has been no AF failure that resulted in exceeding the TS 5.5.12 limit of 0.6 L_a (105,000 sccm) and demonstrates a history of successful tests through the fall of 2013. However, during the PBAPS, Unit 2 fall 2014 refueling outage, planned local leak rate testing identified a condition involving higher than allowable through-seat leakage of two redundant feedwater system check valves (28A and 96A). Pursuant to 10 CFR 50.72(b)(3)(ii)(A), this issue was reported to the NRC on October 29, 2014, as a non-compliance with maximum allowable primary containment leakage rate (L_a). This condition was entered in the plant corrective action program, and the results of the investigation were reported to the NRC as a Licensee Event Report (LER) on December 5, 2014 (ADAMS Accession No. ML14342B002). The LER noted that the cause of the event was determined to be due to operational wear that resulted in a misalignment between the check valve disc and the seat. The LER further stated that, previous to this event, these check valves had demonstrated good operational history and that there were no previous LERs identified involving a failure of redundant check valves resulting in exceeding the L_a primary containment leakage limit.

The licensee noted that prior to this occurrence, the AF minimum pathway summations represent the generally solid performance of the maintenance of Type B and Type C tested components while the AL maximum pathway summations represent the effective management of the Containment Leakage Rate Testing Program by the program owner.

As noted above, the testing during the PBAPS, Unit 2 fall refueling outage identified conditions for two valves that resulted in AF leakage exceeding the TS 5.5.12 leakage limit. Since the last test did not meet the TS limit of $\leq$ 0.6 L_a, the NEI 94-01, Revision 3-A guidelines regarding two consecutive successful tests was not met. However, this appears to be an isolated case given the successful test results from 2000 through 2013. Following corrective actions to repair these valves, the Type C local leak rate test was successfully performed as discussed in the LER. Based on the above, the NRC staff concludes that the results of the Type B and C leak rate tests provide reasonable assurance that local leakage will be maintained below the design-basis leak rate, consistent with the requirements in TS 5.5.12.

3.2.3 Operating Experience

For the PBAPS primary containments, there are four issues related to degradation, or potential degradation, which were identified by the licensee in Section 3.5 of Attachment 1 to the LAR. The issues are as follows:

- LER 2-06-03
- Generic Letter (GL) 87-05
- Through-Wall Torus Shell Crack at James A. FitzPatrick Nuclear Power Plant
- NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing"

Each issue is discussed below.

LER 2-06-03

On October 7, 2006, at 1802 hours, an Unusual Event was declared for PBAPS, Unit 2, as a result of the discovery of a leak at an elbow for piping that penetrates the Primary Containment Suppression Pool (i.e., Torus). The affected piping (High Pressure Coolant Injection (HPCI)/Reactor Core Isolation Cooling (RCIC) Torus Flush line) is in a 4-inch line. This line is normally isolated from the HPCI/RCIC systems by a closed motor-operated valve and is only used during testing activities. Therefore, there was no impact on HPCI or RCIC system functional capability.

The leak was discovered by an equipment operator at approximately 1741 hours during a planned inspection associated with an RCIC system check valve. The leak occurred on the intrados of a 45-degree elbow of the 4-inch piping, and the elbow was located approximately 1 foot above the Torus penetration (i.e., the leak was outside of Primary Containment). The licensee determined the cause of the crack in the elbow was due to cavitation and abrasive erosion and/or localized water-jet cutting resulting from excessively high flow velocities through this piping during test conditions in conjunction with an apparent lack of fusion between the weld backing ring and the weld root at the elbow weld.

The 4-inch carbon steel piping is attached to the Torus and is not isolable from the Torus (i.e., Primary Containment). The piping terminates under the normal Torus water level, and thus, the water in the Torus serves as another barrier to prevent radioactive gaseous releases from the Torus air space during design-basis events. Therefore, there were no actual gaseous releases involved with this event.

The HPCI/RCIC flush line is pressurized during ILRTs. The leakage would have been detected during this test. The last ILRT was successfully completed on October 4, 2000, and there was no leakage identified at that time. Further examination of the leaking elbow noted that axial and circumferential cracking existed at the elbow intrados. Failure analyses of the elbow determined that only minimal leakage existed at the elbow with the as-found indications. This minimal leakage only occurred when the HPCI or RCIC system was being operated in the test mode involving return flow being routed to the Torus.

In the unlikely event that a worst-case design-basis event had occurred and the elbow did not maintain its integrity, additional leakage would have occurred. If both subsystems of containment cooling (including containment spray) were used during the design event, the Torus water level would only be minimally impacted. If only one subsystem of containment cooling were used with no containment spray, then water leakage would have occurred until the HPCI/RCIC flush line became uncovered (approximately 5 feet below normal Torus water level), and a gaseous release could have occurred. The water leakage would be contained within the

Torus Room, and the gaseous leakage would be processed through the secondary containment and Standby Gas Treatment System.

The corrective actions taken by the licensee included replacing the leaking elbow and performing non-destructive tests on the pipe. The similar pipe on Unit 3 was examined and no significant concerns were noted. Extensive walkdowns of similar piping that is attached to the Torus was conducted for both Units 2 and 3, and there were no similar deficiencies discovered. Selected ultrasonic testing was performed on Units 2 and 3 Torus attached piping that involved higher flow rates. These examinations also did not identify any similar concerns.

As a result of this event, PBAPS test procedures were revised to prevent using the HPCI/RCIC Torus Flush line at high flow conditions.

GL 87-05 Request for Additional Information - Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells

GL 87-05 described drywell shell degradation that occurred at the Oyster Creek Nuclear Generating Station as a result of water intrusion into the air gap between the outer drywell surface and the surrounding concrete and subsequent wetting of the sand cushion at the bottom of the air gap. The initial response to this GL for PBAPS was provided by letter to the NRC dated May 11, 1987.

The cause of this degradation was determined to be from water entering the drywell air gap region and becoming trapped in the sand cushion region at the base of the air gap. The air gap region surrounds the outside surface of the drywell and extends from the sand cushion region at the bottom, to just below the drywell bellows region at the top. During refueling activities, a potential leakage path could exist through the drywell bellows region as experienced on the reported Mark I containment. The drywell bellows provide a flexible seal between the drywell and the reactor cavity. The drywell to concrete seal drains are also located in this bellows area. Leakage of these components could allow water to enter the air gap region. However, such water intrusion is not considered credible at PBAPS in that any leakage through the drywell bellows is normally channeled to seal rupture collection pipes and is alarmed in the main control room (MCR).

The licensee described the PBAPS design as incorporating an 8-inch pipe to divert potential drywell bellows leakage to a waste collection tank. This 8-inch drain line is fed by four, 4-inch seal rupture drains equally spaced around the reactor cavity. A flow of 10 gallons per minute through the 8-inch drain line will result in an annunciator alarm on the refueling floor panel and also will result in an alarm in the MCR. Functionality of the alarm and flow switch is verified periodically. Further, unlike the design at the Mark I containment that reported leakage, the PBAPS reactor cavity seal drain line design incorporates full penetration welds instead of bolted connections. Additionally, the PBAPS design incorporates a weir wall that prevents drywell bellows leakage from entering the drywell air gap before being drained away by the seal rupture drains.

The PBAPS design also prevents in-leakage to the sand cushion by use of a sheet metal cover, which is sealed to the drywell shell. This sealed cover separates the sand cushion from the air gap region. Located above the sealed cover plate are an additional four 4-inch air gap drains that drain any in-leakage away from the sealed cover plate.

Additionally, as part of the PBAPS Primary Containment Inservice Inspection Program, several examinations and tests of components associated with the drywell air gap region confirm that abnormal conditions, which could lead to containment degradation, do not exist. These examinations and tests are discussed below.

The following examinations are performed on the four drywell air gap drain lines:

1. A functional test (i.e., smoke test) is performed on the four drywell air gap drains once every 10-year interval to verify that the drywell air gap drain lines are unclogged and functional. The test also verifies that the drain lines are free of water.

2. A visual examination is performed on the drywell air gap drain lines once each period when the refueling cavity is flooded to look for signs of leakage.

The licensee has determined that the above-described examinations and tests have been routinely performed with acceptable results for both units.

Additionally, when stabilizer access hatches (penetrations N-110A through N-110H) are opened to perform the Examination Category E-A examinations on the weld to the shear lugs attached to the exterior of the drywell shell at elevation 194 feet, 8 inches, a VT-3 visual examination is performed on the following items:

1. The drywell exterior stabilizer support.

2. The accessible exterior surface of the drywell to look for evidence of degradation or leakage.

3. The accessible drywell air gap to look for items that could trap water in the unlikely event of leakage through the refueling bellows.

The licensee has determined that, to date, the results of these examinations confirm that no evidence of moisture or degradation exists.

Based on review of the information provided in Section 3.5 of Attachment 1 to the LAR as described above, the NRC staff finds that based on the PBAPS design, along with the monitoring and testing measures described above, there is substantial defense against water entering the drywell air gap region. Therefore, there is reasonable assurance that potential degradation on the outside surface of the drywell will be prevented.

Through-Wall Torus Shell Crack at James A. FitzPatrick Nuclear Power Plant

A through-wall Torus shell crack was discovered at the James A. FitzPatrick Nuclear Power Plant (JAF) on June 27, 2005. Exelon reviewed the issue for applicability to PBAPS and documented the results in the corrective action program.

The JAF HPCI turbine exhaust line that discharges into the suppression pool is open-ended and does not have an end cap or a sparger. The licensee determined that, for PBAPS, its system configurations would not introduce the type of event that occurred at JAF, since the HPCI system design employs the use of a sparger on the turbine exhaust line. The licensee performed VT-2 and VT-3 inspections on the nozzle and the Torus shell next to the HPCI and RCIC exhaust penetrations and the support legs to the Torus shell, which displayed satisfactory results. Therefore, no further actions were required.

NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing"

NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," was issued to alert licensees to problems with local leak rate testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

There are two categories of primary containment bellows at PBAPS:

- Bellows on the vent lines between the drywell and the torus
- Bellows on various drywell pipe penetrations

The bellows listed in PBAPS Updated Final Safety Analysis Report (UFSAR), Table 5.2.2, "Containment Penetrations Compliance with 10 CFR 50, Appendix J," are testable bellows and are tested in accordance with 10 CFR Part 50, Appendix J, Option B, Type B testing. Until Option B was adopted, Type B testing was performed every two years.

Since that time, the test frequency has been extended to once every 6 years. The licensee reviewed the records since June 1977 and determined that there have been no failures of these bellows leakage tests. Additionally, the licensee determined that LLRT procedures for containment expansion bellows include verification of flow through the annulus between plys of the bellows, which ensures that restrictions between the plys that could conceal a leakage path do not exist. Based on the above, the NRC staff concludes that the licensee has adequately addressed the operating experience associated with Information Notice 92-20.

Operating Experience Conclusion

Based on the above, the NRC staff concludes that the licensee has taken appropriate actions in response to operating experience related to degradation, or potential degradation, of the primary containments at PBAPS, Units 2 and 3.

3.2.4 Containment Inspections

Containment Inservice Inspection (CISI) Program

In Section 3.4.2 of Attachment 1 to the LAR, the licensee described the PBAPS CISI program. The licensee stated that the second 10-year CISI interval for PBAPS, Units 2 and 3, began on November 5, 2008, and will end November 4, 2018. The effective edition and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, is the 2001 Edition through the 2003 addenda.

Subsections IWE and IWL of ASME Section XI, contain inservice inspection and repair and replacement rules for metal containment vessels (Class MC) and concrete containment vessels (Class CC), respectively. The reactor containments at PBAPS are free-standing structural steel containments, to which only the requirements of Subsection IWE apply.

As discussed in the licensee's letter dated April 13, 2015, 100% IWE inspections are performed per ASME Section XI every inservice inspection period. Full IWE inspections were performed in 2014 for Unit 2 and in 2013 for Unit 3. The licensee described the results from recent IWE inspections, including any correction actions taken (e.g., repairs). No significant damage or degradation was noted.

Inspection of Concrete Components

PBAPS UFSAR, Section 12.2.1, states that (1) the foundation of the reactor building consists of a monolithic concrete mat supported on sound rock; and (2) this foundation mat also supports the primary containment and its internals, including the reactor vessel pedestal. Furthermore, PBAPS UFSAR, Appendix Q, Section Q.1.16, states that the PBAPS structural monitoring program complies with 10 CFR 50.65 and utilizes visual inspections in managing aging effects for concrete and grout in accessible areas. In response to an NRC staff request for additional information (RAI) relative to PBAPS operating experience and inspection of concrete components (Reference 9), by letter dated April 13, 2015, the licensee stated that the PBAPS Structural Monitoring Program requires inspection of plant structural features in the scope of the Maintenance Rule on a 4-year frequency. This encompasses reinforced concrete throughout the plant, including the reactor building bottom floor slab and reactor pedestal. Since the start of the program in 1997, the inspection findings for both Units 2 and 3 have been limited; and no issues having significant structural impact have been noted. None of the collected data or identified deficiencies is concluded to indicate the potential for degradation of concrete in inaccessible areas, nor in the reactor building base mat or reactor pedestal. No significant damage or degradation has been noted in interior monolithic concrete structures.

Containment Inspection Conclusion

Based on the results of the PBAPS recent IWE inspections and inspections of concrete components discussed above, the NRC staff finds that there has been no evidence to date of significant degradation of PBAPS, Units 2 and 3, primary containments, and that the degradations noted have been entered into the PBAPS corrective action program and appropriately managed and/or corrected. Based on the above evaluation, the NRC staff finds that there is reasonable assurance that the licensee is adequately monitoring and managing age-related degradation of the PBAPS primary containment.

3.3 NRC Staff Evaluation of the Conditions and Limitations

As discussed in SE Section 2.0, in accordance with the guidance in NEI 94-01, Revision 2-A, the licensee proposes to extend the containment Type A test interval from the current approved 10 years to 15 years, based on acceptable performance. The NRC staff's evaluation of the proposed LAR against the limitations and conditions in NEI 94-01, Revision 2-A, is discussed below in SE Section 3.3.1.

As also discussed in SE Section 2.0, in accordance with the guidance in NEI 94-01, Revision 3-A, the licensee proposes to extend the containment Type C test interval from the current approved 60 months to 75 months, with a permissible extension period of 9 months (total of 84 months) for non-routine emergent conditions, based on acceptable performance. The NRC staff's evaluation of the LAR against the limitations and conditions in NEI 94-01, Revision 3-A, is discussed below in SE Section 3.3.2.

3.3.1 NRC Conditions in NEI 94-01, Revision 2-A

In Section 4.1 of the NRC staff SE, incorporated in topical report NEI 94-01, Revision 2-A (Reference 4), the staff concluded that the guidance in the topical report is acceptable for reference by licensees proposing to amend their TSs to permanently extend the ILRT surveillance interval to 15 years, provided that six conditions were satisfied. The NRC staff evaluated whether the licensee addressed and satisfied these conditions for PBAPS, as applicable, in the LAR as discussed below.

a. NRC Condition 1

NRC Condition 1 states: "For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1)."

The licensee states in Section 3.7.1 of Attachment 1 to the LAR that it will utilize the definition in NEI 94-01, Revision 3-A, Section 5.0. This approach is acceptable because the definition remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01. Therefore, the NRC staff finds that the licensee has addressed and satisfied NRC Condition 1.

b. NRC Condition 2

NRC Condition 2 states: "The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3)."

NEI 94-01, Section 9.2.3.2, "Supplemental Inspection Requirements," states that in order to provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval of the Type A test is extended to 15 years.

In response to an NRC staff RAI, by letter dated April 13, 2015, the licensee provided the following information:

a) Per ASME Code, Section XI, IWE Table IWE-2500-1, inspection of the containment vessel pressure retaining boundary, accessible surface areas, and moisture barriers is required during each inspection period.

b) The next Type A tests for each unit, under a 15-year frequency, would occur in Unit 2 refueling outage P2R27 (scheduled for 2028) and Unit 3 refueling outage P3R22 (scheduled for 2019). P2R27 is the second outage of the third period of the third CISI interval. PBAPS is currently in the third period of the second CISI interval.

c) Based on the schedule below, four 100 percent IWE inspections will be performed between each Type A test. For Unit 2, a full IWE inspection will be performed in either P2R26 or in P2R27 refueling outage prior to initiation of the Type A test. For Unit 3, a full IWE inspection will be scheduled under the CISI program for P3R22 refueling outage. This ensures a fourth complete inspection.

IWE Inspection Interval	Period	Unit 2 Refueling Outage(s)	Unit 3 Refueling Outage(s)
1st Interval	3rd	P2R16, P2R17	*P3R15, P3R16
2nd Interval	1st	P2R18, P2R19	P3R17, P3R18
	2nd	*P2R20	P3R19
	3rd	P2R21, P2R22	P3R20, P3R21
3rd Interval	1st	P2R23, P2R24	*P3R22, P3R23
	2nd	P2R25	P3R24
	3rd	P2R26, *P2R27	P3R25, P3R26

*Indicates Type A test performed/scheduled

The licensee's schedule of general visual examinations of accessible containment vessel surfaces results in at least three examinations between Type A tests and one examination immediately prior to the Type A test. This meets the requirements of the proposed revision to TS 5.5.12; the inspection requirements of ASME Code, Section XI, subsection IWE; and NEI 94-01 Revision 3-A, Sections 9.2.1 and 9.2.3.2. The approach to use NEI 94-01, Revision 3-A, is acceptable because Sections 9.2.1 and 9.2.3.2 are identical in both revisions and the licensee has submitted a schedule of inspections to be performed prior to and between Type A Tests. Therefore, the NRC staff finds that the licensee has addressed and satisfied NRC Condition 2.

c. NRC Condition 3

NRC Condition 3 states: "The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3)."

The licensee states that it will continue to perform general visual observations of the accessible interior and exterior surfaces of the containment structure in accordance with containment structural integrity test procedures to meet the requirements of the proposed revision to TS 5.5.12; the inspection requirements of ASME Code, Section XI, subsection IWE; and NEI 94-01, Revision 3-A, Sections 9.2.1 and 9.2.3.2. The approach to use NEI 94-01, Revision 3-A, is acceptable because Sections 9.2.1 and 9.2.3.2 are identical in both revisions and address containment structure areas that are potentially subject to degradation. Therefore, the NRC staff finds that the licensee has addressed and satisfied NRC Condition 3.

d. NRC Condition 4

NRC Condition 4 states: "The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4)."

The licensee indicated in Section 3.7.1 of Attachment 1 to the LAR, that no major modifications to the containment structure are planned. Therefore, the NRC staff finds that the licensee has addressed and satisfied NRC Condition 4.

e. NRC Condition 5

NRC Condition 5 states: "The normal Type A test interval should be less than 15 years.[2] If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2)."

The previous Type A ILRTs were performed in December 2014 (PBAPS, Unit 2) and in October 2005 (PBAPS, Unit 3). The licensee stated in its submittal dated April 13, 2015, that the next Type A ILRTs would be performed in 2028 (PBAPS, Unit 2) and 2019 (PBAPS, Unit 3).

Extending the ILRT interval beyond 15 years does not apply to this LAR, as the licensee only submitted a request for an extension up to 15 years. However, in the event that an extension beyond 15 years is desired, the licensee would need prior NRC approval via a license amendment request consistent with the staff position in Regulatory Issue Summary (RIS) 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50" (Reference 10).

Based on the above, the NRC staff finds that the licensee has addressed and satisfied the intent of the applicable portion of NRC Condition 5 because it proposes a test interval of up to 15 years.

f. NRC Condition 6

NRC Condition 6 states: "For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI [Electric Power Research Institute] Report No. 1009325, Revision 2, including the use of past containment ILRT data."

This condition is not applicable to PBAPS, Units 2 and 3, since they were not licensed under 10 CFR Part 52.

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[2] Although NRC Condition 5 states that the normal Type A test interval should be less than 15 years, the NRC approved the use of Revision 2 to extend Type A test intervals up to 15 years, provided that the conditions are satisfied, as described in Reference 9, Sections 4.1 and 5.0.

3.3.2 NRC Conditions in NEI 94-01, Revision 3-A

In Section 4.0 of the NRC staff SE, incorporated in topical report NEI 94-01, Revision 3-A (Reference 6), the staff concluded that the guidance in the topical report is acceptable for reference by licensees in the implementation for the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J, provided that two conditions were satisfied. The NRC staff has evaluated whether the licensee addressed and satisfied these conditions, for PBAPS, as applicable, in the LAR as discussed below.

a. NRC Condition 1

NRC Condition 1 states, in part, that:

The staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g. BWR MSIVs [boiling-water reactor main steam isolation valves]), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months.

The licensee stated that its post-outage report, as required by Section 12.1 of NEI 94-01, Revision 3-A, will include the margin between the Type B and Type C minimum pathway leak rate summation value, as adjusted, to include the estimate of applicable Type C leakage understatement and its regulatory limit of 0.60 L_a.

The licensee will complete an analysis and determine the appropriate corrective action plan when the potential leakage understatement adjusted Type B and Type C minimum pathway leak rate total is greater than the PBAPS Maintenance Rule leakage summation limit of 0.50 L_a and less than the regulatory limit of 0.60 L_a. The corrective action plan shall focus on the components that have contributed the most to the increase in the leakage summation value and the manner of timely corrective action, as deemed appropriate, that best focus on the prevention of future component leakage performance issues.

Consistent with the generic approval in NEI 94-01, Revision 3-A, the licensee stated that it will only utilize the 9-month grace period beyond 75 months to eligible Type C components for non-routine emergent conditions as specified in Reference 10. These occurrences will be documented in the record of tests.

Based on the above, the NRC staff finds that the licensee has addressed and satisfied NRC Condition 1.

b. NRC Condition 2

NRC Condition 2 states, in part, that:

"When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations."

In response to NRC Condition 2, the licensee states that it will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the AL leakage total for each Type C component on the 75-month extended test interval. The adjustment factor of 1.25 was chosen because the change from a 60-month extended test interval to a 75-month interval is a change of 25 percent. The result is a combined conservative Type C total for all 75-month local leak rate tests being carried forward. The adjustment factor will be included whenever the total leakage summation is required to be updated. An analysis and corrective action plan will be prepared when the summation of the potential leakage understatement adjusted leak rate total for Type C components being tested on a 75-month extended interval and the total of the Type B tested components when the summation is greater than the PBAPS Maintenance Rule limit of 0.50 L_a, but less than the regulatory limit of 0.60 L_a.

If the potential leakage understatement adjusted minimum pathway leak rate is less than the 0.50 L_a administrative leakage summation limit, then the extension for all affected Type C components has been adequately demonstrated, and the calculated local leak rate total represents the actual leakage potential of the penetrations.

The licensee states that, "An adverse trend is defined as three consecutive increases in the final pre-reactor coolant system Mode change Type B and Type C minimum pathway leak rate summation value adjusted to include the estimate of applicable Type C leakage understatement, as expressed in terms of L_a."

The licensee states that it will develop a corrective action plan in the event that an adverse trend is observed. The plan will focus on the components that have contributed the most to the adverse trend in the leakage summation value.

Based on its review of the licensee's submittal, the NRC staff finds that the primary containment leakage rate testing program contains provisions for trending and monitoring that conservatively apply a leakage understatement factor to account for the extended interval. In addition, the post-outage report would contain the necessary information. Therefore, the NRC staff finds that the licensee has addressed and satisfied NRC Condition 2.

3.4 Probabilistic Risk Assessment

3.4.1 Background

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 3-A, states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01 states that the assessment should be performed using the approach and methodology described in EPRI Technical Report 1009325, Revision 2-A[3], "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (Reference 11). The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

In the SE dated June 25, 2008, the NRC staff found the methodology in EPRI Technical Report 1009325, Revision 2, acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. These conditions, set forth in Section 4.2, "Limitations and Conditions for EPRI Report No. 1009325, Revision 2," of the SE for EPRI Technical Report 1009325, Revision 2, stipulate that:

1. The licensee submits documentation indicating that the technical adequacy of its PRA [probabilistic risk assessment] is consistent with the requirements of RG 1.200 ["An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (References 12 and 13)] relevant to the ILRT extension application.

2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small and consistent with the clarification provided in Section 3.2.4.6[4], "Acceptance Guidelines," of the SE for EPRI TR-1009325, Revision 2.

3. The methodology in EPRI TR-1009325, Revision 2, is acceptable provided that the average leak rate for the pre-existing containment large leak accident case (i.e., accident case 3b) used by licensees is assigned a value of 100 times the maximum allowable leakage rate (L_a) instead of 35 L_a.

4. An LAR is required in instances where containment over-pressure is relied upon for ECCS [emergency core cooling system] performance.

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[3] It should be noted that EPRI TR-1009325, Revision 2-A, is also identified as EPRI TR-1018243. This report is publicly available and can be found at www.epri.com by typing "1018243" in the search field box.

[4] The SE for EPRI TR-1009325, Revision 2, indicates that the clarification regarding small increases in risk is provided in Section 3.2.4.5; however, the clarification is actually provided in Section 3.2.4.6.

3.4.2 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval from 10 years to 15 years. The risk analysis was provided in Attachment 3 of the LAR dated November 7, 2014. Additional information was provided by the licensee in its letter dated April 13, 2015, in response to NRC RAIs.

In Section 1.1, "Purpose," of Attachment 3 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in:

- NEI 94-01, Revision 3-A;
- The methodology described in EPRI TR-1009325, Revision 2-A;
- The methodology outlined in EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994; and
- The NRC regulatory guidance outlined in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 14).

In Section 4.4, "Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage," of Attachment 3 to the LAR, the licensee stated that the liner corrosion issue was incorporated into the ILRT extension risk evaluation utilizing the Calvert Cliffs Nuclear Power Plant (CCNPP) methodology and considered applicable liner corrosion events through December 2009. In its April 13, 2015, response to APLA-RAI 2, the licensee discussed corrosion sensitivity results obtained from the CCNPP methodology considering applicable liner corrosion events through December 2014 and confirmed that the conclusions of the analysis provided in Attachment 3 to the LAR remained unchanged.

The licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2, which are described in Section 4.2 of the associated SE. A summary of how each condition has been met is provided in the sections below.

3.4.2.1 Technical Adequacy of the PRA

The first condition stipulates that the licensee submits documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

In RIS 2007-06, "Regulatory Guide 1.200 Implementation" (Reference 15), the NRC clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 (Reference 12) to assess the technical adequacy of the PRA used to support risk-informed applications. Revision 2 of RG 1.200 (Reference 13) will be used for all risk-informed application received after March 2010. In Section 3.2.4.1, "Quality of the PRA," of the SE for EPRI TR-1009325, Revision 2, the NRC staff states that Capability Category I of the ASME and American Nuclear Society (ANS) PRA standard RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," shall be applied as the standard for assessing PRA quality for IRLT extension applications, as approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

Internal Events

In Appendix A, "PRA Technical Adequacy," of Attachment 3 to the LAR, the licensee discussed the technical adequacy of the internal events PRA. In November 2010, the Boiling Water Reactor Owners Group (BWROG) performed a peer review of the PBAPS, Units 2 and 3, internal events PRA models, including internal flooding events. The peer review was performed using ASME/ANS Standard RA-Sa-2009 and RG 1.200, Revision 2. The peer review identified 16 Facts and Observations (F&Os) associated with supporting requirements that were assessed as not meeting Capability Category II, as shown in Table A-1 of Appendix A of Attachment 3 to the LAR. The peer review also identified seven other F&Os that were related to the current ASME/ANS PRA Standard for internal events and internal flooding associated with supporting requirements, which are otherwise met at Capability Category II, as shown in Table A-2 of Appendix A of Attachment 3 to the LAR. The NRC staff concludes that all F&O findings associated with the 2010 PBAPS, Units 2 and 3, PRA models were properly assessed and dispositioned in regard to this application.

In support of the LAR for the PBAPS, Units 2 and 3, Extended Power Uprate (EPU), dated September 28, 2012, the licensee developed an EPU PRA model from the 2010 PBAPS, Unit 2, PRA. The EPU PRA model accounts for EPU conditions, including CDF scenarios for a pre-existing leak from containment and operators failing to align the appropriate cross-ties in sufficient time. Based on the SE for the PBAPS, Units 2 and 3, EPU, dated August 25, 2014 (ADAMS Accession No. ML14133A046), the NRC staff approved the EPU LAR and concluded that the licensee had adequately modeled in the EPU PRA the potential impacts associated with the implementation of the EPU. The EPU plant modifications were completed on Unit 2 following the 2014 refueling outage, and will be implemented on Unit 3 during the fall 2015 refueling outage. The licensee used the EPU PRA model to support this LAR for a permanent extension of the Type A containment ILRT frequency for PBAPS, Units 2 and 3.

Based on the above assessment of the quality of the 2010 PBAPS, Unit 2, PRA model and the NRC staff's assessment of the quality of the EPU PRA model, dated August 25, 2014, and given the level of PRA quality needed to support a permanent extension of the Type A containment ILRT frequency, the staff concludes that the EPU PRA model is of sufficient quality to evaluate risk associated with the permanent extension of the Type A containment ILRT frequency for PBAPS, Units 2 and 3.

External Events

In Section 3.2.4.2, "Scope of the PRA," of the SE for EPRI TR-1009325, Revision 2, the NRC staff stated that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT

> intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

In Section 5.7, "External Events Contribution," of Attachment 3 to the LAR, the licensee performed an analysis of the external events contributions. The results of the PBAPS Individual Plant Examination of External Events (IPEEE) are documented in the PBAPS IPEEE Main Report, dated May 1996. The primary areas of external event evaluation at PBAPS were internal fire and seismic events. Based on the SE of the IPEEE, dated November 22, 1999, the NRC staff concludes that the IPEEE results are reasonable given the PBAPS design, operation, and history; and therefore, the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities. However, there are no comprehensive CDF and LERF values available from the IPEEE to support this LAR risk assessment.

To calculate CDF for internal fire, the licensee developed a fire PRA model that is based on the PBAPS IPEEE fire analysis using the EPRI Fire-Induced Vulnerability Evaluation method. The fire PRA model was updated in 2007 to include analysis of the MCR and cable spreading room. This fire PRA model has been used in previously submitted and approved analyses (e.g., SE for the PBAPS EPU LAR, dated August 25, 2014; SE for the PBAPS one-time extension of the Type A ILRT frequency, dated July 20, 2010). Bounding seismic CDF values for PBAPS, Units 2 and 3, have been developed and reported in NRC Generic Issue 199, dated August 2010, which is based on the updated 2008 U.S. Geological Survey seismic hazard curves. Using the internal fire and seismic event CDFs, the licensee estimated the risk increases (see Section 3.2.2 of this LAR) using an approach consistent with that in Section 5.2.5.2, "Potential Impacts from External Events," of EPRI TR-1009325, Revision 2-A. Therefore, the NRC staff finds that the information used to estimate the risk increases associated with extending the Type A containment ILRT frequency due to external events is acceptable.

Based on the above discussion of internal and external events, the NRC staff finds that the first condition is met.

3.4.3 Estimated Risk Increase

The second condition stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small, and is consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.6, "Acceptance Guidelines," of the NRC SE for EPRI TR-1009325, Revision 2. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Additionally, for plants that rely on containment over-pressure for net positive suction head (NPSH) for ECCS injection, both CDF and LERF, will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. Thus, the associated risk metrics include CDF, LERF, population dose, and CCFP.

Details of the risk assessment are provided in Attachment 3 to the LAR. The risk impacts are reported for a change in the Type A ILRT test interval from a three per 10 years (the test frequency under 10 CFR Part 50 Appendix J, Option A) to a once per 15 years (risk impact from baseline) and from a once per 10 years to a once per 15 years (risk impact from current). The licensee reported the results of the plant-specific risk assessment in Section 5.7.5 of Attachment 3 to the LAR. In its April 13, 2015, response to APLA-RAI 1, the licensee addressed the NRC staff's concern that the external events multiplier in Table 5.7-1 in Attachment 3 to the LAR could be non-conservative and potentially underestimate the risk results. The licensee appropriately updated the external events multiplier to be more reflective of the fire and seismic accident sequences. In addition, updated risk results were provided in the response (i.e., total LERF in Table 1-3 and ΔLERF, Δperson-roentgen equivalent man (rem)/year, and ΔCCFP in Table 1-2 of the response).

The following conclusions are drawn from the licensee's analysis associated with extending the Type A containment ILRT frequency:

1. The reported increase in LERF for a change in test frequency from three tests in 10 years to one test in 15 years is 7.74E-07 per year (Table 1-2 in the response to APLA-RAI 1, dated April 13, 2015). This value includes both internal and external events (i.e., internal fires and seismic events) and the impacts from corrosion. This increase in internal and external events risk is considered to be "small" (i.e., between 1E-07 per year and 1E-06 per year) per acceptance guidelines in RG 1.174. According to RG 1.174, an assessment of the baseline LERF is required to show that the total LERF is less than 1E-05 per reactor year. The licensee estimated the new total LERF to be 6.72E-06 per year (Table 1-3 in the response to APLA-RAI 1, dated April 13, 2015), which is below the total LERF value of 1E-05 per reactor year in RG 1.174.

2. The increase in CCFP due to a change in test frequency from three tests in 10 years to one test in 15 years is 1.02 percent (Table 1-2 in the response to APLA-RAI 1, dated April 13, 2015). This is less than the acceptance guideline value of 1.5 percent for a small increase in CCFP, as provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SE for EPRI TR-1009325, Revision 2.

3. Given a change in Type A ILRT frequency from three tests in 10 years to one test in 15 years, the reported increase in the total population dose is 1.23 person-rem per year or 0.52 percent of the total population dose (Table 1-2 in the response to APLA-RAI 1, dated April 13, 2015). The increase in population dose is less than the values associated with a small increase, as provided in EPRI TR-1009325, Revision 2-A.

Based on the risk assessment results, the NRC staff concludes that the increase in LERF is small and consistent with the acceptance guidelines of RG 1.174. In addition, the increase in the total integrated plant risk and the magnitude of the change in the CCFP for the requested change are small and supportive of the LAR. The defense-in-depth philosophy is maintained because the independence of barriers will not be degraded as a result of the requested change, and the use of the quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the NRC staff finds that the second condition is met.

3.4.4 Leak Rate for the Large Pre-existing Containment Leak Rate Case

The third condition stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2, acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.

As noted by the licensee in Section 1.3, "Acceptance Criteria," of Attachment 3 to the LAR, the methodology in EPRI TR-1009325, Revision 2-A, incorporates the use of 100 L_a as the average leak rate for the pre-existing containment large leak rate accident case, and this value has been used in the plant-specific risk assessment. Accordingly, the NRC staff finds that the third condition is met.

3.4.5 Applicability if Containment Over-Pressure is Credited for ECCS Performance

The fourth condition stipulates that in instances where containment over-pressure is relied upon for ECCS performance, an LAR is required to be submitted. In Section 5.8, "Containment Overpressure Impacts on CDF," of Attachment 3 to the LAR, the licensee stated that the plant modifications made in support of the EPU described in the associated LAR, dated September 28, 2012 and approved by the NRC in the SE dated August 25, 2014, eliminated reliance on containment accident pressure to provide adequate NPSH margin. However, this relies on the successful alignment of the RHR System heat exchanger cross-tie and the High Pressure Service Water System cross-tie under certain accident sequences. Therefore, the licensee estimated the bounding increase in the internal events CDF associated with the ILRT interval extension to be 8E-9 per year. The external events contribution (i.e., internal fire and seismic events) can be estimated by multiplying the increase in the internal events CDF (i.e., 8E-9 per year) by the external event bounding multiplier (i.e., 19.5) given in the April 13, 2015, response to APLA-RAI 1. As such, the bounding increase in total CDF associated with the ILRT interval extension is 1.6E-7 per year, which is considered to be very small when compared with the risk acceptance guidelines of RG 1.174. Accordingly, the NRC staff finds that the fourth condition is met.

3.4.6 PRA Conclusion

Based on the above discussions, the NRC staff concludes that the LAR for a permanent extension of the Type A containment ILRT frequency from once in 10 years to once in 15 years for PBAPS, Units 2 and 3, is acceptable.

3.5 Evaluation of Proposed Changes to TS 5.5.12

Replacement of Reference to RG 1.163

The licensee proposed to remove the reference to RG 1.163 and replace it with a reference to NEI 94-01, Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008. The replacement would become the implementation documents used by the licensee to implement the PBAPS, Units 2 and 3, performance-based leakage testing program in accordance with Option B of 10 CFR Part 50, Appendix J.

TS 5.5.12 currently states, in part:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, 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 exceptions to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J:

The proposed TS 5.5.12 would state, in part:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exception:

Based on the NRC staff's evaluation in this SE, the licensee has demonstrated that the proposed test intervals are acceptable and meet the applicable conditions and limitations in NEI 94-01, Revisions 2-A and 3-A. Therefore, the NRC staff concludes that the above change is acceptable.

Deletion of One Time Exceptions Previously Granted

The licensee proposed to remove the following exceptions from the TS 5.5.12

> Unit 2:
>
> b. Section 9.2.3: The first Type A test performed after the October 2000 Type A test shall be performed no later than October 2015.

Unit 3:

b. Section 9.2.3: The first Type A test performed after the December, 1991 Type A test shall be performed no later than December, 2006.

The above exceptions relate to previous NRC approvals of one-time extensions of the ILRT Type A test frequencies. These exceptions are no longer necessary since the proposed LAR would allow a 15-year ILRT test interval on a permanent basis. As such, the NRC staff concludes that the proposed deletions are acceptable.

3.6 Technical Evaluation Conclusion

Based on the discussion in SE Sections 3.1 through 3.5, the NRC staff concludes that the proposed amendments are acceptable.

4.0 STATE CONSULTATION

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

5.0 PUBLIC COMMENTS

On January 20, 2015, the NRC staff published its regular biweekly notice in the *Federal Register* regarding "Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations" (80 FR 2747). With respect to amendments proposed to be issued, in accordance with the requirements in 10 CFR 50.91(a)(2)(ii), the notice provided a 30-day period for public comment on the staff's proposed determination that the associated amendments do not involve a significant hazards consideration. The notice included the staff's proposed no significant hazards consideration (NSHC) determinations for several different proposed amendments, including the proposed amendments for PBAPS, Units 2 and 3.

One comment was received in response to the *Federal Register* notice (ADAMS Accession No. ML15027A337). The comment raised concerns regarding the public availability of the Final Safety Analysis Reports for nuclear power plants in general. The comment did not cite to any of the specific proposed amendments included in the *Federal Register* notice. In addition, the issues discussed in the comment do not specifically pertain to the proposed NSHC determination for any of the proposed amendments included in the *Federal Register* notice. As such, the NRC staff is not providing a response to the comment in this SE.

6.0 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. 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

(80 FR 2749). 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.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Exelon Generation Company, LLC, "License Amendment Request - Revise Technical Specifications Section 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies," Docket Nos. 50-277 and 50-278, November 7, 2014 (ADAMS Accession No. ML14315A084).

2. Exelon Generation Company, LLC, "Response to Request for Additional Information - License Amendment Request to Revise Technical Specifications Section 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies," Docket Nos. 50-277 and 50-278, April 13, 2015 (ADAMS Accession No. ML15104A361).

3. U.S. Nuclear Regulatory Commission, RG 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (ADAMS Accession No. ML003740058).

4. NEI TR NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008 (ADAMS Accession No. ML100620847).

5. U.S. Nuclear Regulatory Commission, Final Safety Evaluation, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,' and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, 'Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," June 25, 2008 (ADAMS Accession No. ML081140105).

6. NEI TR NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012 (ADAMS Accession No. ML12221A202).

7. Deleted.

8. U.S. Nuclear Regulatory Commission, Final Safety Evaluation, "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01, Revision 3, 'Industry Guideline for

Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,' (TAC No. ME2164)," June 8, 2012 (ADAMS Accession No. ML121030286).

9. U.S. Nuclear Regulatory Commission, "Peach Bottom Atomic Power Station, Units 2 and 3, Draft Request for Additional Information (TAC Nos. MF5172 and MF5173)," March 19, 2015 (ADAMS Accession No. ML15078A083).

10. U.S. Nuclear Regulatory Commission, Regulatory Issue Summary 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50," December 8, 2008 (ADAMS Accession No. ML080020394).

11. Electric Power Research Institute, TR-1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Final Report," August 2007 (ADAMS Accession No. ML072970208).

12. U.S. Nuclear Regulatory Commission, RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007 (ADAMS Accession No. ML070240001).

13. U.S. Nuclear Regulatory Commission, RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014).

14. U.S. Nuclear Regulatory Commission, RG 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (ADAMS Accession No. ML100910006).

15. U.S. Nuclear Regulatory Commission, Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," March 22, 2007 (ADAMS Accession No. ML070650428).

16. Exelon Generation Company, LLC, "License Amendment Request to Revise Technical Specifications Section 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies" Docket Nos. 50-277 and 50-278, August 10, 2015 (ADAMS Accession No. ML15222B171).

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T. Hilsmeier
T. Lamb
R. Ennis

Date: September 8, 2015

September 8, 2015

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS RE: EXTENSION OF TYPE A AND TYPE C LEAK RATE TEST FREQUENCIES (TAC NOS. MF5172 AND MF5173)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment Nos. 302 and 306 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3. These amendments consist of changes to the Technical Specifications (TSs) and Facility Operating Licenses in response to your application dated November 7, 2014, as supplemented by letters dated April 13, 2015, and August 10, 2015.

The amendments revise the TSs associated with the primary containment leakage rate testing program. Specifically, the amendments extend the frequencies for performance of the Type A containment integrated leakage rate test and the Type C containment isolation valve leakage rate test, which are required by Title 10 of the *Code of Federal Regulations* Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

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

Sincerely,
/RA/

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

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 302 to Renewed DPR-44
2. Amendment No. 306 to Renewed DPR-56
3. Safety Evaluation

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RidsNrrDssStsb Resource
RidsNrrDeEmcb Resource
RidsNrrDraApla Resource
THilsmeier, NRR
DWoodyatt, NRR
BLee, NRR
FFarzam, NRR
RidsNrrLALRonewicz Resource

ADAMS Accession No.: ML15196A559

OFFICE	DORL/LPL1-2/PM	DORL/LPL1-2/LA	DORL/LPL1-2/LA	DE/EMCB/BC(A)	DSS/SCVB/BC
NAME	TLamb	REnnis	LRonewicz	YLi	RDennig
DATE	7/21/15	9/8/15	7/16/15	8/11/15	8/11/15
OFFICE	DRA/APLA/BC	OGC	DORL/LPL1-2/BC	DORL/LPL1-2/PM	
NAME	SRosenberg	JWachutka	DBroaddus	REnnis	
DATE	8/11/15	8/14/15	9/8/15	9/8/15	

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