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

April 29, 2016

Mr. Dennis L. Koehl
President and CEO/CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REQUEST TO EXTEND THE 10-YEAR CONTAINMENT INTEGRATED LEAK RATE TEST FREQUENCY TO 15 YEARS (CAC NOS. MF6176 AND MF6177)

Dear Mr. Koehl:

The Commission has issued the enclosed Amendment No. 210 to Facility Operating License No. NPF-76 and Amendment No. 197 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 29, 2015, as supplemented by letters dated June 29, October 8, and November 11, 2015, and March 17, 2016.

The amendments change TS 6.8.3.j, "Containment Leakage Rate Testing Program," to allow a permanent extension of the Type A primary containment integrated leak rate testing frequency from once every 10 years to once every 15 years.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

FOR

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 210 to NPF-76
2. Amendment No. 197 to NPF-80
3. Safety Evaluation

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 210
License No. NPF-76

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

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)*, acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated April 29, 2015, as supplemented by letters dated June 29, October 8, and November 11, 2015, and March 17, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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 license 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.

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*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 210, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-76 and the Technical Specifications

Date of Issuance: April 29, 2016

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 197
License No. NPF-80

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

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)*, acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated April 29, 2015, as supplemented by letters dated June 29, October 8, and November 11, 2015, and March 17, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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 license 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.

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*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-80 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 197, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-80 and the Technical Specifications

Date of Issuance: April 29, 2016

ATTACHMENT TO LICENSE AMENDMENT NOS. 210 AND 197

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

DOCKET NOS. 50-498 AND 50-499

Replace the following pages of the Facility Operating Licenses, Nos. NPF-76 and NPF-80, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-76

REMOVE	INSERT
-4-	-4-

Facility Operating License No. NPF-80

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT
6-9	6-9

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 210, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SER)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Safety Parameter Display System (Section 18, SSER No. 4)*

Before startup after the first refueling outage, HL&P[**] shall perform the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to issues as described in Section 18.2 of SER Supplement 4.

(6) Supplementary Containment Purge Isolation (Section 11.5, SSER No. 4)

HL&P shall provide, prior to startup from the first refueling outage, control room indication of the normal and supplemental containment purge sample line isolation valve position.

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

** The original licensee authorized to possess, use and operate the facility was HL&P. Consequently, historical references to certain obligations of HL&P remain in the license conditions.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 197 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SR)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) License Transfer

Texas Genco, LP shall provide decommissioning funding assurance, to be held in decommissioning trusts for South Texas Project, Unit 2 (Unit 2) upon the direct transfer of the Unit 2 license to Texas Genco, LP, in an amount equal to or greater than the balance in the Unit 2 decommissioning trust immediately prior to the transfer. In addition, Texas Genco, LP shall ensure that all contractual arrangements referred to in the application for approval of the transfer of the Unit 2 license to Texas Genco, LP to obtain necessary decommissioning funds for Unit 2 through a non-bypassable charge are executed and will be maintained until the decommissioning trusts are fully funded, or shall ensure that other mechanisms that provide equivalent assurance of decommissioning funding in accordance with the Commission's regulations are maintained.

(6) License Transfer

The master decommissioning trust agreement for Unit 2, at the time the direct transfer of Unit 2 to Texas Genco, LP is effected and thereafter, is subject to the following:

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* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

6.8.3.g (continued)

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR 190.

h. Not Used

i. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all based on applicable ASTM Standards. The purpose of the program is to establish the following:

1) Acceptability of new fuel oil prior to addition to the diesel generator fuel oil storage tanks by determining that the fuel oil has:

   a. an API gravity or absolute specific gravity within limits,

   b. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   c. a clear and bright appearance with proper color;

2) Within 31 days following addition of new fuel oil to the diesel generator fuel oil storage tanks, verify that the properties of the new fuel oil, other than those addressed in 6.8.3.i.1 above, are within limits for ASTM 2D fuel oil; and

3) Total particulate concentration of fuel oil is $\leq$ 10 mg/l when tested every 31 days using a test method based on ASTM D-2276.

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

j. Containment Leakage Rate Testing Program

A program shall be established to implement 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 Nuclear Energy Institute (NEI) topical report NEI 94-01 Revision 2-A, dated October 2008.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 210 AND 197 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By application dated April 29, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15128A352), as supplemented by letters dated June 29, October 8, and November 11, 2015, and March 17, 2016 (ADAMS Accession Nos. ML15198A147, ML15293A509, ML15329A304, and ML16089A406, respectively), STP Nuclear Operating Company (STPNOC, the licensee) requested changes to the Technical Specifications (TSs) for South Texas Project (STP), Units 1 and 2. The supplemental letters dated October 8 and November 11, 2015, and March 17, 2016, 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) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 14, 2015 (80 FR 48924).[1]

The license amendment request (LAR) proposes changes to Appendix A, TSs, to allow a permanent extension of the 10-year frequency of the Type A Integrated Leak Rate Test (ILRT) that is required by TS 6.8.3.j, "Containment Leakage Rate Testing Program," to 15 years. In particular, the LAR proposes to replace the reference to NRC Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Rate Testing Program," September 1995 (ADAMS Accession No. ML003740058), with a reference to a topical report by the Nuclear Energy Institute (NEI) NEI 94-01, "Industry Guideline for Implementing the Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," Revision 2-A, October 2008 (ADAMS Accession No. ML100620847), as the implementation document used for the performance-based leakage testing program.

[1] The *Federal Register* notice was corrected on August 20, 2015 (80 FR 50663).

2.0 BACKGROUND

On August 13, 1996, the NRC approved Amendment No. 84 for STP Unit 1, and Amendment No. 71 for STP Unit 2, to Facility Operating License Nos. NPF-76 and NPF-80, respectively, authorizing the implementation of 10 CFR Part 50, Appendix J, Option B for Type A, B, and C tests (ADAMS Accession No. ML021300572).

STPNOC previously submitted an LAR for STP Unit 1 and Unit 2 to extend the ILRT intervals on a one-time basis from 10 years to 15 years in a letter dated August 2, 2001 (ADAMS Accession No. ML012250197). This one-time extension was approved by the NRC, as Amendment Nos. 143 and 131 to Facility Operating License Nos. NPF-76 and NPF-80, respectively, on September 17, 2002 (ADAMS Accession No. ML022410163).

The purpose of this LAR is to extend on a permanent basis the 10-year frequency of the Type A primary containment ILRT intervals to 15 years. STPNOC requested approval of the proposed license amendment in an appropriate time, to support the scheduling of the STP Unit 2 refueling outage in fall 2016. Based on the currently required frequency of 10 years, the next STP Unit 1 containment ILRT performance is due during fall 2019.

3.0 REGULATORY EVALUATION

Section 50.54(o) of 10 CFR requires that the primary reactor containments for water-cooled power reactors are subject to the requirements set forth in Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The regulations under 10 CFR Part 50, Appendix J include two options: "Option A – Prescriptive Requirements," and "Option B – Performance-Based Requirements," either of which will meet the requirements of the appendix. These testing requirements ensure that (a) leakage through containments or systems and components penetrating containments does not exceed allowable leakage rates specified in the TS, and (b) the integrity of the containment structure is maintained during the service life of the containment. Both STP Unit 1 and Unit 2 have voluntarily adopted and implemented Option B to meet the requirements of 10 CFR Part 50, Appendix J.

Option B specifies performance-based requirements and criteria for preoperational and subsequent leakage-rate testing. These requirements are met by performance of Type A tests to measure the containment system overall integrated leakage rate, Type B pneumatic tests to detect and measure local leakage rates across pressure-retaining leakage-limiting boundaries such as penetrations, and Type C pneumatic tests to measure containment isolation valve (CIV) 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 integrity of the overall containment system as a barrier to fission product release.

Currently, STP's TS 6.8.3.j requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in RG 1.163 which endorses, with certain exemptions, NEI 94-01, Revision 0, dated July 21, 1995 (ADAMS Accession No. ML11327A025).

Section 50.55a, "Codes and standards," of 10 CFR contains the Containment In-Service Inspection (CISI) requirements that, in conjunction with the requirements of Appendix J, ensure the continued leak-tight and structural integrity of the containment during its service life.

The regulations under 10 CFR 50.65(a), "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," state, in part, that the licensee:

> ...shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience.

The Type A test is an overall ILRT of the containment structure. NEI 94-01, Revision 0, specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months, but this "should be used only in cases where refueling schedules have been changed to accommodate other factors." Amendment Nos. 143 and 137 for STP Units 1 and 2, respectively, allowed a one-time extension of the ILRT interval to 15 years. However, the long-term ILRT test interval requirement in TS 6.8.3.j remained at 10 years.

The history of results of the integrated leakage rate (L_a) Type A testing for STP was provided in Section 4.3 of the LAR. In Note 1 to Table 4.3-1 of Section 4.3, the licensee stated that all ILRTs have been performed at peak-containment pressure, P_a, following a loss-of-coolant accident (LOCA) in the plant TSs in effect at the time of the test. The definition of the Type A test pressure is consistent with the definition of P_a. All Type A tests were successful, in that the test results were less than 1.0 L_a and less than the TS 6.8.3.j.1 limiting values. Both P_a and L_a limiting values are defined in TS 6.8.3.j.

Guidance for extending Type A ILRT surveillance intervals beyond 10 years is provided in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008.

The Type A, Type B, and Type C test results must not exceed the L_a with margin, as specified in TS 6.8.3.j. Option B also requires a general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration that may affect the containment leak-tight integrity. This inspection must be conducted prior to each Type A test at a periodic interval between tests based on the performance of the containment system.

STPNOC proposes to extend the STP Unit 1 and Unit 2 intervals for the primary containment ILRT to 15 years from the last ILRT. The last Unit 1 ILRT was completed on October 3, 2009, and the last Unit 2 ILRT was completed on March 28, 2007. The ILRTs for each unit are currently required to be performed at a frequency of once every 10 years. Therefore, the next Unit 1 ILRT is due by October 2019 and the next Unit 2 ILRT is due by March 2017. Since the next refueling outage for Unit 2 is scheduled for fall 2016, the licensee requested approval of the proposed amendment in time to support planning of the Unit 2 fall 2016 outage. Using the

proposed interval of 15 years, the next STP Unit 1 ILRT will be due by October 2024, and the next STP Unit 2 ILRT will be due by March 2022.

The regulations under 10 CFR 50, Appendix J, Option B, Section V.B.3, require that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage-testing program must be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the NRC in endorsed guidance.

4.0 TECHNICAL EVALUATION

4.1 Licensee's Proposed Changes

The current TS 6.8.3.j, "Containment Leakage Rate Testing Program," states:

> 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-Testing Program," dated September 1995. The current ten-year interval between performance of the integrated leak rate (Type A) test, beginning September 24, 1991, for Unit 2 and March 10, 1995, for Unit 1, has been extended to 15 years (a one-time change).

The proposed amendment revises TS 6.8.3.j to replace the reference to RG 1.163 with a reference to NEI 94-01, Revision 2-A, and delete the last sentence of the paragraph. The proposed change will revise the first paragraph of TS 6.8.3.j to state:

> 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 Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 2-A, dated October 2008.

4.2 NRC Staff Evaluation

The proposed changes in the LAR would revise the aforementioned portion of STP TS 6.8.3.j by replacing the reference to RG 1.163 with a reference to NEI 94-01, Revision 2-A, and the conditions and limitations specified therein, as the implementation document used by STP to implement the Unit 1 and Unit 2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. Consistent with the guidance contained in both NEI 94-01, Revision 2-A, the licensee justified the proposed changes by demonstrating adequate performance of the STP containments based on (a) the historical plant-specific containment leakage testing program results; (b) the CISI program results; and (c) a STP plant-specific risk and probabilistic assessment. The NRC staff reviewed the LAR and supplements based on deterministic considerations with regard to containment leak-tight integrity. A probabilistic risk assessment (PRA) was also performed, as discussed in Section 4.3 of this SE.

4.2.1 Description of the Primary Containments

In the LAR dated April 29, 2015, the licensee described the primary containments for both Units 1 and 2:

> The Containment is a fully continuous, steel-lined, post-tensioned, reinforced concrete structure consisting of a vertical cylinder with a hemispherical dome, supported on a flat foundation mat. The cylinder and dome are post-tensioned with high-strength unbounded wire tendons.
>
> The Containment wall is independent of the adjacent interior and exterior structures, with sufficient space provided between the Containment wall and the adjacent structures to prevent contact under all loading conditions.
>
> A continuous welded steel liner plate is provided on the entire inside face of the Containment to limit the release of radioactive materials into the environment. The nominal thickness of the liner in the wall and dome is 3/8 inch. A 3/8-inch-thick plate is used on top of the foundation mat and is covered with a 24 in. concrete fill slab.
>
> An increased plate thickness up to 2 in. is provided around all penetrations and for the crane girder brackets.
>
> Leak-chase channels and angles are provided at the bottom of liner seams, which, after construction, are inaccessible for leaktightness examination due to the 2-ft interior, fill slab.

STP TS 6.8.3.j indicates that the containments were designed for a leakage rate, L_a, not to exceed 0.3 percent of containment air weight per day at the calculated peak pressure, P_a. TS 6.8.3.j states that the peak calculated containment internal pressure for the design basis LOCA, P_a, is 41.2 pounds per square inch gauge (psig).

4.2.2 Integrated Leakage Rate Test, Type A Test Frequencies

As required by 10 CFR 50.54(o), the STP containments are subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the STP Appendix J Testing Program Plans are based on RG 1.163, which endorses NEI 94-01, Revision 0. The LAR proposes to revise the Appendix J Testing Program Plan by implementing the guidance contained in NEI 94-01, Revision 2-A, and the conditions and limitations contained therein.

By letter dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC published a safety evaluation (SE), with limitations and conditions, for NEI 94-01, Revision 2. In the SE, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate

testing, subject to the limitations and conditions noted in Section 4.0 of the NRC's SE. Section 4.1 of the SE establishes limitations and conditions pertaining to deterministic requirements, while Section 4.2 establishes limitations and conditions pertaining to the plant's PRA analysis. More explicitly, the SE included provisions for extending the ILRT Type A interval to a maximum of 15 years subject to the six limitations and conditions provided in the SE. The NRC noted in the SE that NEI 94-01, Revision 2, incorporates the regulatory positions stated in RG 1.163. The accepted version of NEI 94-01, Revision 2, was subsequently issued as Revision 2-A. The NEI issued Revision 2-A to NEI 94-01 on November 19, 2008. With Revision 2-A, the RG was revised to incorporate the NRC staff's SE.

LAR Table 4.10-1, "NEI 94-01, Revision 2-A, Limitations and Conditions," indicates that STP will meet the limitations and conditions in Section 4.1 of the NRC's SE for NEI 94-01, Revision 2-A.

The leakage rate testing requirements of 10 CFR 50 Appendix J Option B and the CISI requirements mandated by 10 CFR 50.55a, together, ensure the continued leak-tight and structural integrity of the containment during its service life.

Type B testing ensures that the leakage rate of individual containment penetration components is acceptable. Type C testing ensures that individual CIVs are essentially leak tight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

In the LAR, the licensee proposes to invoke NEI 94-01, Revision 2-A, along with the conditions and limitations contained therein as the reference document for STP in TS 6.8.3.j. The licensee is not applying to extend the frequencies of the Type B or Type C performance-based test intervals beyond 120 or 60 months, respectively.

The NRC staff has found that the use of NEI 94-01, Revision 2-A, is acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT surveillance interval to 15 years, provided the following six conditions are satisfied, as applicable.

Condition 1

> For calculating the Type A leakage rate, the licensee should use the definition in the NEI 94-01, Revision 2-A, instead of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002 (Refer to SE Section 3.1.1.1).

<u>STPNOC LAR Compliance Statement</u> (as stated)

> STPNOC will utilize the definition in NEI 94-01, Revision 2-A, Section 5.0.

NRC Staff Assessment

Section 3.2.9, "Type A test performance criterion," of ANSI/ANS-56.8-2002[2] defines the "performance leakage rate" and states, in part:

> The performance criterion for a Type A test is met if the performance leakage rate is less than L_a. The performance leakage rate is equal to the sum of the measured Type A test UCL and the total as-left MNPLR of all Type B or Type C pathways isolated during performance of the Type A test.

The NRC staff's SE Section 3.1.1.1, for NEI 94-01, Revision 2, states, in part:

> Section 5.0 of NEI [Topical Report (TR)] 94-01, Revision 2, uses a definition of "performance leakage rate" for Type A tests that is different from that of ANSI/ANS-56.8-2002... The definition contained in NEI TR 94-01, Revision 2, is more inclusive because it considers excessive leakage in the performance determination. In defining the minimum pathway leakage rate, NEI TR 94-01, Revision 2, includes the leakage rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position prior to the performance of the Type A test. Additionally, the NEI TR 94-01, Revision 2, definition of performance leakage rate requires consideration of the leakage pathways that were isolated during performance of the test because of excessive leakage in the performance determination. The NRC staff finds this modification of the definition of "performance leakage rate" used for Type A tests to be acceptable.

Section 5.0 of NEI 94-01, Revision 2-A, reads:

> The **performance leakage rate** is calculated as the sum of the Type A upper confidence limit (UCL) and as-left minimum pathway leakage rate (MNPLR) leakage rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position (i.e., drained and vented to containment atmosphere) prior to performing the Type A test. In addition, leakage pathways that were isolated during performance of the test because of excessive leakage must be factored into the performance determination. The performance criterion for Type A tests is a performance leak rate of less than $1.0L_a$.

The NRC staff reviewed the definitions of "performance leakage rate" contained NEI 94-01, Revision 2, and Revision 2-A. The staff concluded that the definitions contained in the two revisions are identical. Based on this, the staff agrees with the licensee's response that "*STPNOC will utilize the definition in NEI 94-01, Revision 2-A, Section 5.0.*"

Based on the above review, the NRC staff finds that the licensee has adequately addressed Condition 1.

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[2] American National Standards Institute /American Nuclear Society (ANSI/ANS)-56-8-2002, Reaffirmed August 9, 2011, "Containment System Leaking Testing Requirements."

Condition 2

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).

STPNOC Compliance Statement

Inservice Inspection Program for Concrete Containment

In the LAR, the licensee stated that the Code of record for the third 10-year interval examination of the concrete containment (IWL) components, including related requirements, for Units 1 and 2 is the 2004 Edition with no addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Division 1 in accordance with 10 CFR 50.55a(b)(2)(vi). The licensee also stated that the inservice inspection schedule shall be at 1, 3, and 5 years following the completion of the containment structural integrity test and every 5 years thereafter as required by IWL-2400 as indicated in the Table 2 below:

Table 2
STP Unit 1 and 2, IWL Concrete and Tendon Examination Schedule

10th Year Examination - 1998
20th Year Examination scheduled 2008, performed 2009
25th Year Examination scheduled 2013, performed 2014
30th Year Examination - 2018
35th Year Examination - 2023
40th Year Examination - 2028

Inservice Inspection Program for Containment Metal Liner - IWE

In the LAR, the licensee stated that the Code of record for the second 10-year interval examination of containment metal liner (IWE) components, including related requirements, for Units 1 and 2 is the 2004 Edition with no addenda of the ASME Code, Section XI, Division 1 in accordance with 10 CFR 50.55a(b)(2). The additional requirements specified by 10 CFR 50.55a(b)(2)(ix) are identified in the program procedure.

The schedules for the second and third IWE Intervals are shown in the Table 3 and Table 4 below:

Table 3
STP Unit 1, IWE Containment Metal Liner Examination Schedule

Interval	Period	Dates	Outage	Dates
2	1	9/9/2009 – 9/8/2013	1RE16	04/03 - 04/27/2011
			1RE17	10/20 - 11/19/2012
	2	9/9/2013 – 9/8/2016	1RE18	03/29 - 04/20/2014
			1RE19	10/03 - 11/12/2015
	3	9/9/2016 – 9/8/2019	1RE20	04/01 - 04/23/2017
			1RE21	10/07 - 11/04/2018
3	1	9/9/2019 – 9/8/2023	1RE22	03/2020[a]
			1RE23	10/2021[a]
	2	9/9/2023 – 9/8/2026	1RE24	03/2023[a]
			1RE25	10/2024[a]
	3	9/9/2026 – 9/8/2029	1RE26	03/2026[a]
			1RE27	10/2027[a]

[a]Outage dates are approximations. Exact dates and outage durations have yet to be placed into the long-range outage plan.

Table 4
STP Unit 2, IWE Containment Metal Liner Examination Schedule

Interval	Period	Dates	Outage	Dates
2	1	9/9/2009 - 09/8/2013	2RE14	03/27 – 05/02/2010
			2RE15	10/29 – 11/22/2011
			2RE16	04/27 – 05/24/2013
	2	9/9/2013 – 9/8/2016	2RE17	10/04 – 10/29/2014
			2RE18	04/02 – 04/28/2016
	3	9/9/2016 – 9/8/2019	2RE19	09/30 – 10/24/2017
			2RE20	03/31 – 04/22/2020
3	1	9/9/2019 – 9/8/2023	2RE21	10/2021[a]
			2RE22	03/2023[a]
			2RE23	10/2024[a]
	2	9/9/2023 – 9/8/2026	2RE24	03/2026[a]
			2RE25	10/2027[a]
	3	9/9/2026 – 9/8/2029	2RE26	03/2029[a]
			2RE27	10/2030[a]

[a]Outage dates are approximations. Exact dates and outage durations have yet to be placed into the long-range outage plan.

NRC Staff Assessment

The NRC staff's SE Section 3.1.1.3, for NEI 94-01, Revision 2, states:

> ...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 for the Type A test has been extended to 15 years. NEI TR 94-01, Revision 2, recommends that these inspections be performed in conjunction or coordinated with the examinations required by ASME Code, Section XI, Subsections IWE and IWL. The NRC staff finds that these visual examination provisions, which are consistent with the provisions of regulatory position C.3 of RG 1.163, are acceptable considering the longer 15 year interval. Regulatory Position C.3 of RG 1.163 recommends that such examination be performed at least two more times in the period of 10 years. The NRC staff agrees that as the Type A test interval is changed to 15 years, the schedule of visual inspections should also be revised. Section 9.2.3.2 in NEI TR 94-01, Revision 2, addresses the supplemental inspection requirements that are acceptable to the NRC staff.

NEI 94-01, Revision 2-A, Section 9.2.3.2, "Supplemental Inspection Requirements," states:

> 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 for the Type A test has been extended to 15 years. It is recommended that these inspections be performed in conjunction or coordinated with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL required examinations.

The NRC staff reviewed LAR Attachment 1, Section 4.1, "Inspections." Subsections 4.1.3, "Inservice Inspection Program for Concrete Containment – IWL," and 4.1.4, "Inservice Inspection Program for Containment Metal liner – IWE," provide details of past and future inspection schedules. The referenced sections of the LAR contain:

a) Table 4.1.3-1, "STP Unit 1 and 2, IWL Concrete and Tendon Examination Schedule";

b) Table 4.1.4-1, "STP Unit 1, IWE Containment Metal Liner Examination Schedule"; and

c) Table 4.1.4-2, "STP Unit 2, IWE Containment Metal Liner Examination Schedule."

These tables provide a summary of past STP containment IWL/IWE supplemental inspection schedules, including those planned in the future. Table 4.1.3-1 lists the IWL inspections scheduled for the time period from 1998 through 2028. Tables 4.1.4-1 and 4.1.4-2 list the IWE

inspections scheduled for the time period from October 2009 through October 2029. The schedules provided by the licensee for past and future inspections comply with the requirements of Condition 2.

Condition 3

> The licensee addresses the areas of the containment structure potentially subjected to degradation (Refer to SE Section 3.1.3).

STPNOC LAR Compliance Statement (as stated)

> Reference Sections 4.1.3 and 4.1.4 this submittal.

NRC Staff Assessment

The NRC staff's SE Section 3.1.3, for NEI 94-01, Revision 2, states, in part:

> In approving for Type A tests the one-time extension from 10 years to 15 years, the NRC staff has identified areas that need to be specifically addressed during the IWE and IWL inspections including a number of containment pressure-retaining boundary components (e.g., seals and gaskets of mechanical and electrical penetrations, bolting, penetration bellows) and a number of the accessible and inaccessible areas of the containment structures (e.g., moisture barriers, steel shells, and liners backed by concrete, inaccessible areas of ice condenser containments that are potentially subject to corrosion).

Summary statements for the historical IWE/IWL containment inspections are contained in LAR Attachment 1, Section 4.1.5, "Results of Recent IWE/IWL Examinations."

The NRC staff reviewed LAR Attachment 1, Sections 4.1.1, 4.1.2, 4.1.3, and 4.1.4. The information provided in the referenced sections comply with the requirements of Condition 3.

Condition 4

> The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable (Refer to SE Section 3.1.4).

STPNOC LAR Compliance Statement (as stated)

> STP Unit 1 and Unit 2 steam generator and reactor vessel head replacements have been completed. There are no planned modifications for STP Units 1 and 2 that will require a Type A test prior to the next Units 1 and 2 Type A test proposed under this LAR. There is no anticipated addition or removal of plant hardware within the containment building, which could affect its leak-tightness.

NRC Staff Assessment

The NRC staff's SE Section 3.1.4, for NEI 94-01, Revision 2, states, in part:

> Section 9.2.4 of NEI TR 94-01, Revision 2, states that: "Repairs and modifications that affect the containment leakage integrity require LLRT or short duration structural tests as appropriate to provide assurance of containment integrity following the modification or repair. This testing shall be performed prior to returning the containment to operation." Article IWE-5000 of the ASME Code, Section XI, Subsection IWE (up to the 2001 Edition and the 2003 Addenda), would require a Type A test after major repair or modifications to the containment. In general, the NRC staff considers the cutting of a large hole in the containment for replacement of steam generators or reactor vessel heads, replacement of large penetrations, as major repair or modifications to the containment structure.

The licensee's compliance statement with "Condition 4" based on the projection that there are no planned modifications of the STP Unit 1 and Unit 2 containment buildings which could affect the containment's leak tightness is acceptable.

Condition 5

> The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI 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).

STPNOC LAR Compliance Statement (as stated)

> STPNOC will follow the requirements of NEI 94-01, Revision 2-A, Section 9.1. In accordance with the requirements of 94-01, Revision 2-A, [Safety Evaluation Report (SER)] Section 3.1.1.2, STPNOC will also demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15-year interval is required.

NRC Staff Assessment

Section 3.1.1.2, "Deferral of Tests Beyond The 15-Year Interval" of the NRC staff's SE dated June 25, 2008, states:

> As noted above, Section 9.2.3, NEI TR 94-01, Revision 2, states, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history." However, Section 9.1 states that the "required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes." The NRC staff believes that extensions of the performance-based Type A test interval beyond the required

> 15 years should be infrequent and used only for compelling reasons. Therefore, if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01, Revision 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists.

The NRC staff notes that the licensee has acknowledged the requirements of NEI 94-01, Revision 2-A, SER Section 3.1.1.2 and accepted the NRC staff position discussed in Condition 5. The licensee has confirmed its understanding that any extension of the Type A test interval beyond the upper-bound performance-based limit of 15 years should be infrequent and that any requested permission (i.e., for such an extension) will demonstrate to the NRC staff that an unforeseen emergent condition exists.

Based on the above review, the NRC staff finds that the licensee has adequately addressed "Condition 5".

Condition 6

> 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 94-01, Revision 2, and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, including the use of past containment ILRT data.

STPNOC LAR Compliance Statement (as stated)

> Not applicable. STP was not licensed under 10 CFR Part 52.

NRC Staff Assessment

The NRC staff finds that STP Unit 1 and Unit 2 are operating reactors currently licensed per the requirements of 10 CFR Part 50, and therefore, concludes that "Condition 6" does not apply.

Based on the above evaluation of each condition, the NRC staff determined that the licensee has adequately addressed the six conditions identified in Section 4.1 of the NRC SE for NEI 94-01, Revision 2-A. Therefore, the NRC staff finds it acceptable to adopt the "conditions and limitations" of NEI 94-01, Revision 2-A, as part of the implementation documents in TS 6.8.3.j, "Containment Leakage Rate Testing Program."

4.2.3 Licensee's Proposal for Extension of Type A Test Interval up to 15 Years

With the LAR and its supplements, the licensee proposed to extend both STP Unit 1 and Unit 2 current performance-based Type A test intervals to no longer than 15 years by adopting NEI 94-01, Revision 2-A, and the conditions and limitations contained therein as the implementation documents in TS 6.8.3.j. This change would allow STP Unit 1 to conduct the next Type A test by October 2024, in lieu of the current due time of October 2019. This change would allow STP Unit 2 to conduct the next Type A test by March 2022, in lieu of the current due time of March 2017. The licensee justified these proposed changes by demonstrating adequate

performance of the STP Unit 1 and Unit 2 containments based on plant-specific containment leakage testing program results and CISI (IWE/IWL) results and supported by a plant-specific risk assessment, consistent with the guidance in NEI 94-01, Revision 2-A, and the conditions and limitations contained therein. The LAR and its supplements were reviewed and evaluated by the NRC staff, as discussed in this section, based on deterministic considerations, with regard to containment structural and leak-tight integrity, if the current ILRT interval is extended to 15 years. A PRA is provided in Section 4.3 of this SE.

4.2.4 Peak Calculated Containment Internal Pressure

The current calculated peak containment pressure for the limiting case (Double Ended Hot Leg (DEHL) break with minimum safety injection and minimum containment heat removal) is 39.5 psig. The current value of 41.2 psig included in TS 6.8.3.j for the peak calculated containment internal pressure for the design basis loss-of-coolant accident (LOCA), P_a, contains some margin. The licensee noted that for Westinghouse-designed pressurized-water reactors like STP Units 1 and 2, the mass and energy release input due to a LOCA for the calculation of the peak pressure is provided by Westinghouse. Nuclear Safety Advisory Letters (NSALs) are issued by Westinghouse to advise licensees of potential safety issues related to the basic components supplied by Westinghouse, including mass and energy release data for a LOCA inside containment. In Section 4.5.1 of Attachment 1 to the LAR letter dated April 29, 2015, the licensee addressed the impact of NSAL 11-5 and NSAL 14-2 on mass and energy release and the peak calculated pressure. The licensee stated that the changes discussed in the NSALs would result in a change to the peak containment pressure and as a compensatory action revised 0PSP11-ZA-005, "Local Leakage Rate Test Calculations, Guidelines and Program." to require a P_a of 43.2 psig to ensure sufficient margin until the Updated Final Safety Analysis Report (UFSAR) is updated to reflect the corrected value. The licensee determined that there are no operability concerns due to the increased P_a and documented the same in an operability review under CREE 11-12472-1. In response to the NRC staff's Request for Additional Information (RAI)-1 dated September 3, 2015 (ADAMS Accession No. ML15251A216), the licensee provided updated information stating that subsequent to the LAR letter dated April 29, 2015, the analysis has been completed and that the results of the analysis demonstrate that P_a of 41.2 psig in TS 6.8.3.j remains bounding.

In its March 17, 2016, response to a follow-up RAI, the licensee stated that the analysis was performed in accordance with the NRC-approved methodology in WCAP-10325-P-A and it incorporated the issues addressed in NSAL-06-06, NSAL-11-05, and NSAL-14-02.[3] The incorporation of these issues resulted in an increase in mass and energy releases and did not reduce the conservatism of the previous analysis.

The licensee continued in its response that the previous containment pressure–temperature response used the CONTEMPT computer code. The revised containment response analysis

[3] Westinghouse Electric Company, LLC, NSAL-06-06, "LOCA Mass and Energy Release Analysis," dated June 6, 2006; NSAL-11-05, "Westinghouse LOCA Mass and Energy Release Calculation Issues," dated July 25, 2011; and NSAL-14-02, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties," dated July 25, 2011.

was performed with the GOTHIC computer code utilizing the NRC-approved methodology documented in Dominion DOM-NAF-3-NP-1A.[4]

The Containment model, Containment initial conditions, heat sinks, Reactor Containment fan Cooler (RCFC) model, Containment spray model, Sump recirculation model, and the Heat exchanger model used in the GOTHIC model are essentially same as used in the CONTEMPT model.

The licensee made two significant modeling changes when converting from the CONTEMPT to GOTHIC computer models. The first change replaced the Uchida/Tagami condensing heat transfer model with Diffusion Layer Model (DLM) for heat transfer between atmosphere and heat sinks. The second change added the liquid/vapor interface model between the atmosphere and the sump. The CONTEMPT model used a constant heat transfer coefficient, whereas the GOTHIC model used a split heat transfer option that switches the heat transfer from the vapor phase to the liquid phase.

The licensee then stated that the combination of the increased mass and energy release and changes to the containment model for the limiting case (DEHL break with minimum safety injection and minimum containment heat removal) resulted in a change of the peak containment pressure from 39.5 psig to 40.1 psig, which is bounded by the P_a value of 41.2 psig in TS 6.8.3.j.

The revised mass and energy releases, containment pressure response, and Equipment Qualification pressure/temperature response reflecting the revised analysis were incorporated in the STP UFSAR by UFSAR Change Notice 3136.

Westinghouse InfoGram is also a tool, like the NSALs, that Westinghouse employs to keep its customers informed of discoveries involving safety analysis methodologies. InfoGram IG-14-1 dated November 5, 2014,[5] stated that Westinghouse LOCA M&E release methodology was found to use a stainless volumetric heat capacity value that is lower than values currently listed in the ASME Boiler and Pressure Vessel Code. Westinghouse's generic evaluation in IG-14-1 concluded that its current LOCA mass and energy release rate for input to containment calculations is conservative and that its methodology has been proven to be overall conservative for containment integrity and downstream analysis. In its March 17, 2016, response to an NRC staff RAI dated March 1, 2016 (ADAMS Accession No. ML16053A187), the licensee described the plant-specific analysis/evaluations performed to determine the applicability of the finding reported in InfoGram IG-14-1 to STP and its impact on the current analysis.

In its letter dated March 17, 2016, the licensee stated that additional discussions with Westinghouse identified that the stainless steel specific values used in the current analysis are approximately 20 percent lower than the current ASME Boiler and Pressure Vessel Code data. A sensitivity study was performed to assess the impact of additional energy that could be released due to higher specific heat values. The sensitivity study determined the amount of

[4] Dominion, Approval Topical Report DOM-NAF-3 NP-A, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment," dated November 16, 2006 (ADAMS Accession No. ML063190467).

[5] Westinghouse Electric Company, InfoGram IG-14-1, "Material Properties for Loss-of-Coolant Accident Mass and Energy Release Analyses," dated November 14, 2014.

additional energy required to increase the current peak calculated containment pressure of 40.1 psig to exceed the P_a value of 41.2 psig in TS 6.8.3.j. The results of the study showed that energy released from the primary and secondary metal mass would have to increase by 46 percent for P_a to exceed 41.2 psig. Based on the 20 percent lower stainless steel specific values, the additional energy that could be released due to the difference in the specific heat values is estimated to be 20 percent. Therefore, the licensee concluded that margin still exists before the P_a value is exceeded during a LOCA event.

The mass and energy release input for the revised containment-pressure response analysis appropriately addressed the Westinghouse notification of potential errors contained in NSAL-06-06, NSAL-11-05, and NSAL-14-02. The licensee also performed a sensitivity analysis to address the Westinghouse notification of a potential error in mass and energy release contained in InfoGram IG-14-1. The revised containment response analysis was performed with the GOTHIC computer code utilizing the NRC-approved methodology documented in Dominion DOM-NAF-3-NP-A. The revised analysis resulted in a peak calculated containment pressure of 40.1 psig. STPNOC's conclusion that the P_a value of 41.2 psig in the current TS 6.8.3.j remains bounding is supported by the results of the revised analysis.

4.2.5 STP Type A Test Performance History

Per TS 6.8.3.j, STP Unit 1 and Unit 2 were designed for a maximum allowable containment leakage rate L_a of 0.3 percent of containment air weight per day at the calculated peak pressure, P_a.

Since March 1987, a total of four ILRTs have been performed on STP Unit 1, all with satisfactory results. Since October 1998, a total of three ILRTs have been performed on STP Unit 2, all with satisfactory results. All the ILRT test results were documented in LAR Attachment 1, Section 4.1, Table 4.3.1. These test results are reproduced in Table 4.3-1 below:

TABLE 4.3-1, Integrated Leakage Rate Testing History

Unit (Date)	Mass point Leakage (Weight percent/Day)	Acceptance Limit (Weight percent/Day)	Test Pressure[1] (Psig)
1 (03/25/1987)	0.0320	0.225	37.4
1 (01/10/1991)	0.0688	0.225	39.5
1 (03/10/1995)	0.020	0.225	44.5
1 (10/03/2009)	0.1270[2]	0.225	42.4753
2 (09/27/1988)	0.034	0.225	38.3
2 (09/23/1991)	0.0765	0.225	44.6
2 (03/28/2007)	0.144196[2]	0.225	41.606

Note 1: All ILRTs have been performed at Peak Containment Post LOCA pressure as identified in the plant's TSs in effect at the time of the test.

Note 2: The step change in containment leakage recorded in the 2007 and 2009 ILRTs is due in part to the tests being performed in 8 to 9 hours verses the 24 hours performed previously.

The NRC staff notes that the last sentence of Section 9.2.3, "Extended Test Intervals," of NEI 94-01, Revision 2-A, states, "[i]n the event where previous Type A tests were performed at reduced pressure (as described in 10 CFR 50, Appendix J, Option A), at least one of the two consecutive periodic Type A tests shall be performed at peak accident pressure (P_a)."

As described in Table 4.3-1, the last two STP Unit 1 tests were performed in December 1995 and in October 2009. As noted in Note 2 of Table 4.3-1, both tests were performed at a peak containment post-LOCA pressure identified in the plant's TSs in effect at the time of the test. Per TS 6.8.3.j, the peak calculated containment internal pressure for the design basis LOCA, P_a, is 41.2 psig. Therefore, the previous consecutive periodic Type A tests were performed at a test pressure that exceeds the current P_a in TS 6.8.3.j.

As described in Table 4.3-1, the last two STP Unit 2 tests were performed in September 1991 and in September 2007. As noted in Note 2 of Table 4.3-1, both tests were performed at a peak containment post-LOCA pressure identified in the plant's TSs in effect at the time of the test. Per TS 6.8.3.j, the peak calculated containment internal pressure for the design basis LOCA, P_a, is 41.2 psig. Therefore, the previous consecutive periodic Type A tests were performed at a test pressure that exceeds the current P_a in TS 6.8.3.j.

STP Unit 1 and Unit 2 TS 6.8.3.j states, in part:

> The maximum allowable containment leakage rate, L_a, is 0.3 percent of containment air weight per day.
>
> Leakage rate acceptance criteria are:
>
> 1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit start-up following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq .75\ L_a$ as-left and $\leq 1.0\ L_a$ as-found for Type A tests.

As noted in Table 4.10-1, NEI 94-01, Revision 2-A, Limitations and Conditions of Attachment 1 to the LAR, the steam generator and reactor vessel head replacements have been completed for STP Units 1 and 2. The licensee also stated in the table that there are no planned modifications for STP Unit 1 and Unit 2 that will require a Type A test prior to the next Unit 1 and Unit 2 Type A tests proposed under this LAR. Further, there is no anticipated addition or removal of plant hardware within the containment building, which could affect the leak tightness of containment.

The STP Unit 1 and Unit 2 containments were designed for a leakage rate L_a not to exceed 0.3 percent by weight of containment air per day at the calculated peak pressure, P_a. As displayed in Table 4.3-1, STP Unit 1 and Unit 2 have substantial margin to the "As found" performance limit as described in TS 6.8.3.j of L_a equal to 0.3 percent weight/day for all historical ILRTs.

As required by NEI 94-01, Revision 2-A, Section 9.1.2, further extensions in test intervals are contingent upon two consecutive successful periodic Type A tests and the requirements as stated therein in Section 9.2.3. Past STP Unit 1 and Unit 2 ILRT results have confirmed that the containment is acceptable with respect to the design criterion of 0.3 percent leakage of containment air weight (L_a) per day at the design basis LOCA pressure (P_a). Since the last two Type A tests for STP Unit 1 and Unit 2 had "as found" test results of less than 1.0 L_a, a test frequency of 15 years in accordance with NEI 94-01, Revision 2-A, would be acceptable for both STP units.

Based on the above STP Unit 1 and Unit 2 ILRT test results, the NRC staff concludes that the requirements of Sections 9.1.2 and 9.2.3 of NEI 94-01, Revision 2-A, have been satisfied.

4.2.6 Local Leakage Rate Tests: Type B and Type C Tests

As required by 10 CFR 50.54(o), the STP Unit 1 and Unit 2 containments are subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J allows that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, STP Unit 1 and Unit 2 TS 6.8.3.j, "Containment Leakage Rate Testing Program," is implemented in accordance with the guidelines contained in RG 1.163. The licensee's LAR with its supplement proposes to revise the STP Unit 1 and Unit 2 TS 6.8.3.j by replacing Option B implementation document RG 1.163 with NEI 94-01, Revision 2-A, along with the conditions and limitations contained therein to govern the test frequencies and the grace periods for Type A, Type B, and Type C tests.

The leakage rate testing requirements of 10 CFR 50 Appendix J Option B (Type A, Type B, and Type C tests) and the CISI requirements mandated by 10 CFR 50.55a, together, ensure the continued leak-tight and structural integrity of the containment during its service life.

Type B testing ensures that the leakage rate of individual containment penetration components is acceptable. Type C testing ensures that individual CIVs are essentially leak tight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

The licensee's LAR proposes to invoke NEI 94-01, Revision 2-A, as the implementation document for STP Unit 1 and Unit 2 "Containment Leakage Rate Testing Program" TS 6.8.3.j to govern its Type B and Type C local leak-rate test (LLRT) program.

4.2.7 STP Type B Test and Type C Test Performance History

STP Unit 1 and Unit 2 TS 6.8.3.j states, in part:

> The maximum allowable containment leakage rate, L_a, is 0.3 percent of containment air weight per day.
>
> Leakage rate acceptance criteria are:
>
> 1) Containment overall leakage rate acceptance criterion is $\leq$1.0 L_a. During the first unit start-up following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the combined Type B and Type C tests, and $\leq$ .75 L_a as-left and $\leq$1.0 L_a as-found for Type A tests.

The NRC staff reviewed the local leak rate summaries contained in LAR Attachment 1 Table 4.4.1-1, "Unit 1 Type B and C LLRT As-Found/As-Left Trend Summary," and Table 4.4.1-2, "Unit 2 Type B and C LLRT As-Found/As-Left Trend Summary."

The data contained in Table 4.4.1-1 and Table 4.4.1-2 supported the following conclusions:

- The "As-Found" minimum pathway leak rate average for STP Unit 1 shows an average of 4.49 percent of 0.6 L_a with a high of 4.89 percent of 0.6 L_a (i.e., 0.029 L_a).
- The "As-Found" minimum pathway leak rate average for STP Unit 2 shows an average of 7.54 percent of 0.6 L_a with a high of 8.82 percent of 0.6 L_a (i.e., 0.053 L_a).

Based on the review of the data contained in the two tables, the NRC staff concludes that the aggregate results of the "As-Found Min Path" for all the Type B and C tests from 2004 through 2014 demonstrates a history of successful tests since the aggregate test results were significantly less than the Type B and Type C test TS limit of < 0.60 L_a (i.e., 455,050 standard cubic centimeters per minute (sccm)) contained in TS 6.8.3.j.

The NRC staff reviewed the corrective actions identified in LAR Attachment 1 Table 4.4.1-3, "Unit 1 Type C LLRT Program Implementation Review," taken for the valves that exceeded the administrative leakage limits set in the licensee's Containment Leak Rate Testing Program in the most recent Unit 1 Type C LLRT Program tests during the Refueling Outages 1RE17 in 2012 and 1RE18 in 2014. The table did not identify any components that have not demonstrated acceptable performance during the outage 1RE17. A single penetration, M-43, during outage 1RE18 exceeded the maintenance criteria leakage value and the licensee provided the corrective actions taken which resulted in satisfactory post maintenance surveillance. The staff concludes that adequate corrective action had been taken for the Unit 1 valves when leakage values exceeded their administrative values.

The NRC staff reviewed the corrective actions identified in LAR Attachment 1 Table 4.4.1-4, "Unit 2 Type C LLRT Program Implementation Review," taken for valves that exceeded their

administrative values in the most recent Unit 2 Type C, LLRT Program tests during the Refueling Outages 2RE15 in 2011 and 2RE16 in 2013. The table identified two penetrations, M-44 and M-48, with leakage exceeding the administrative limit, all of which were accepted for continued service with leakage still left above the administrative limit but below the maintenance criteria. In its October 8, 2015, response to RAI-1, the licensee stated that additional maintenance was performed during the spring 2015 outage, 2 RE17. The post-maintenance as-left LLRT on M-44 resulted in a satisfactory leak rate that is well below the administrative limit. In regards to M-48, the licensee stated that it tested below administrative limits during the as-found LLRT in outage 2RE17 in 2015 and that M-48 will remain on its nominal 18-month frequency until satisfactory completion of two as-found tests below administrative limits. The licensee has contingency plans for valve repair/replacement if leak rate were to be above administrative limits during outage 2RE18. The staff concludes that adequate corrective actions have been taken or are in place for Unit 2 valves that performed below the set criteria in their LLRTs.

The percentage of the total number of Unit 1 and 2 Type B tested components that are on 120-month extended performance-based test intervals is 90 percent.

The percentage of the total number of Unit 1 and 2 Type C tested components that are on 60-month extended performance-based test intervals is 80 percent.

The NRC staff finds that the total percent of Type B and Type C tested components that are on the maximum test intervals (120 months for Type B and 60 months for Type C) are adequate.

In summary, the NRC staff concludes that:

- in its letter dated October 8, 2015, the licensee adequately responded to the staff's RAI;
- the cumulative Type B and C test results were well below the acceptance limit of TS 6.8.3.j;
- the licensee has a corrective action program that appropriately addresses poor performing valves.

Therefore, the NRC staff finds that the licensee's Type B and Type C leakage rate test program results support and complement the current Type A tests, as required by Option B of 10 CFR 50, Appendix J, including the permanent extension of Type A test to 15 years.

4.2.8 <u>NRC Overall Evaluation of the Proposed Extension of ILRT and LLRT Test Intervals</u>

The NRC staff reviewed the Type A, Type B, and Type C leakage test results related to the licensee's proposal to extend the 10 CFR 50, Appendix J test intervals for Type A tests.

The ILRT results provided in Section 4.3 of Attachment 1 to the LAR indicate that the previous two consecutive Type A tests at STP Unit 1 and Unit 2 were successful with containment performance leakage rates significantly less than the maximum allowable containment leakage rate of 0.30 percent containment air weight per day (1.0 L_a at P_a) and less than the Type A test TS limit of ≤ 0.75 L_a contained in TS 6.8.3.j. Therefore, the NRC staff finds that the performance

history of Type A tests supports extending the current ILRT interval on a permanent basis to 15 years as permitted by NEI 94-01, Revision 2-A, and the conditions and limitations contained therein.

The NRC staff reviewed the local leak rate summaries contained in LAR Attachment 1 Table 4.4.1-1 and Table 4.4.1-2 and notes that the aggregate results of the "As-Found Min Path" for all the recent Type B and C tests are significantly less than the Type B and Type C test TS limit of < 0.60 L_a contained in TS 6.8.3.j. The staff reviewed the corrective actions identified in LAR Attachment 1 Table 4.4.1-3 taken for the valves that exceeded the administrative leakage limits in the most recent Unit 1 Type C LLRT Program tests during the Refueling Outage 1RE18 in 2014 and notes that adequate corrective action for those valves has been performed. The staff reviewed the corrective actions identified in LAR Attachment 1 Table 4.4.1-4 taken for the valves that exceeded the administrative leakage limits in the most recent Unit 2 Type C, LLRT Program tests during the Refueling Outages 2RE15 in 2011 and 2RE16 in 2013, and notes that adequate corrective action for those valves has been performed. Therefore, the NRC staff finds that the licensee is effectively implementing the Type B and Type C leakage rate test program, as required by 10 CFR 50, Appendix J, Option B. Accordingly, the NRC staff finds that the performance history of Type B tests and Type C tests is supportive of the permanent extension of Type A testing frequency to 15 years.

4.2.9 Conclusion

Based on the NRC staff's review of the licensee's submittal of April 29, 2015, as supplemented, and the regulatory and technical evaluations above, the staff finds that there is reasonable assurance that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting NEI 94-01, Revision 2-A, and the conditions and limitations specified therein, as the 10 CFR 50 Appendix J, Option B implementation documents.

The NRC staff also finds that the licensee adequately implemented its Containment Leakage Rate Testing Program (i.e. Type A, B, and C leakage tests), for the STP Unit 1 and Unit 2 containment structures. The results of past ILRTs and recent LLRTs demonstrate acceptable performance of the STP Unit 1 and Unit 2 containments and demonstrate that the structural and leak-tight integrity of the containment structures are being adequately maintained. The staff also finds that the structural and leak-tight integrity of the STP Unit 1 and Unit 2 containments will continue to be monitored and maintained upon STP's adopting NEI 94-01, Revision 2-A, and the conditions and limitations contained therein as the 10 CFR 50 Appendix J, Option B implementation documents. Accordingly, the NRC staff determined that there is reasonable assurance that the structural and leak-tight integrity, for both containments, will continue to be maintained, without undue risk to public health and safety, if the current Type A test intervals are extended to 15 years.

The NRC staff concludes that it is acceptable for STP Unit 1 and Unit 2 to: (i) revise TS 6.8.3.j, "Containment Leakage Rate Testing Requirements," to adopt NEI 94-01, Revision 2-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, as the 10 CFR 50 Appendix J, Option B implementation documents; and (ii) extend, on a permanent basis, the Type A test interval up to 15 years.

4.3 Probabilistic Risk Assessment

4.3.1 Background

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 3-A dated July 2012 (ADAMS Accession No. ML122210202), 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 1018243,[6] "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." 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 an SER dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC staff found the methodology in NEI 94-01, Revision 2, and EPRI TR-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 of the SER for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submit documentation indicating that the technical adequacy of their Probabilistic Risk Assessment (PRA) is consistent with the requirements of 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), relevant to the ILRT extension application. Additional application specific guidance on the technical adequacy of a PRA used to extend ILRT intervals is provided in the SER for EPRI TR-1009325, Revision 2.

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[7] of the SER for EPRI TR-1009325, Revision 2.

3. The methodology in EPRI TR-1009325, Revision 2, is acceptable provided 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 emergency core cooling system (ECCS) performance.

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

[7] Section 4.2 of the SER 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.

4.3.2 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval to once in 15 years. The risk assessment was provided in Attachment 4 to the LAR dated April 29, 2015, as supplemented by letter dated June 29, 2015. Additional information was provided by the licensee in response to NRC RAIs in its letter dated October 8, 2015.

In Section 4.11.1 of Attachment 1 to the LAR, the licensee stated that the plant-specific risk assessment uses the guidance in NEI 94-01, Revision 3-A[8] (ADAMS Accession No. ML12221A202); the methodology described in EPRI TR-1018243, Revision 2-A of 1009325; and the NRC regulatory guidance outlined in 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). Additionally, the licensee used the methodology from Calvert Cliffs Nuclear Power Plant to assess the risk from undetected containment leaks due to steel liner corrosion.

The licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2, which are listed in Section 4.2 of the NRC SER of June 25, 2008. A summary of how each condition has been met is provided in the sections below.

4.3.3 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, Revision 2, relevant to the ILRT extension application.

Consistent with the information provided in Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428), the NRC staff uses Revision 2 of RG 1.200 to assess technical adequacy of the PRA used to support risk-informed applications received after March 2010. In Section 3.2.4.1 of the SER for NEI 94-01, Revision 2 and EPRI TR-1009325, Revision 2, the NRC staff stated that Capability Category I of the ASME PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

Section 4.11.2 of Attachment 1 to the LAR states that the PRA model, Revision 7.2 was used for the ILRT extension application. STP's PRA is a full-scope level 1 and level 2 PRA that includes both internal and external initiating events, including fires, floods, seismic events, and high winds. The PRA is maintained and updated under a PRA configuration control program in accordance with station procedures. Periodic reviews are conducted and updates are performed, if necessary, for plant changes including performance data, procedures, and modifications. The reviews and updates are performed by qualified personnel with independent reviews and approvals.

[8] NEI 94-01, Revision 3-A, added guidance for extending Type C Local Leak Rate Test (LLRT) surveillance intervals beyond 60 months. The guidance for extending Type A ILRT surveillance intervals beyond 10 years is the same as that in Revision 2-A.

The LAR states that STP's plant-specific PRA was peer reviewed against RG 1.200, Revision 1, January 2007 (ADAMS Accession No. ML070240001). During the review of Risk-Managed Technical Specifications (RMTS) for STP Units 1 and 2 (SE dated July 13, 2007; ADAMS Accession No. ML071780186), the NRC staff determined that the licensee's internal events PRA satisfies the guidance of RG 1.200, Revision 1, at Capability Category II. As described in its supplement dated June 29, 2015, the licensee performed a gap assessment and concluded that the internal events PRA meets the requirements of RG 1.200, Revision 2, with the exception of the technical elements related to uncertainty analysis in internal flooding (supporting requirements IFSO-B3, IFSN-B3, IFEV-B3, and IFQU-B3). The licensee stated that an internal flooding uncertainty analysis was not performed, as required by the PRA standard, because all internal flooding sequences have low frequency and have been screened out. The licensee clarified that the low frequency of flooding scenarios is due to the plant's design with three safety trains and high degree of compartmentalization. The licensee further stated in its October 8, 2015, supplement that there is no quantification of internal flooding sequences in the licensee's PRA model (because all sequences were screened out) and, therefore, the internal flooding initiating events do not contribute to plant's total core damage frequency. Given that the licensee's evaluation of internal flooding sequences showed a low frequency for all those sequences in STP, the NRC staff finds that not performing uncertainty analysis for internal floods has no significant impact on the risk metrics for the ILRT extension application. Therefore, based on NRC's review of STP's internal events PRA performed for the RMTS application, and on the gap assessment provided by the licensee against RG 1.200, Revision 2, the staff concludes that the quality of the licensee's internal events PRA is acceptable for the ILRT extension application.

In Section 3.2.4.2 of the SER for NEI 94-01, Revision 2 and 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."

As stated in the LAR and clarified in the supplement dated June 29, 2015, the scope of STP's PRA includes internal hazards, including internal floods, and external hazards, such as internal fires, seismic events, high winds, and external flooding. The risk metrics reported in the LAR include the impact from these external hazards. As part of the review of the RMTS application (SE dated July 13, 2007), the NRC staff reviewed the internal fire PRA model and found it addresses the technical characteristics and attributes required per RG 1.200, Revision 1, and is

appropriate to support RMTS. As part of the same application, the staff also reviewed STP's seismic, high wind, and external flooding PRA models, and concluded that STP PRA external events models are acceptable to support the RMTS application. Based on NRC's review of STP's external events PRA for the RMTS application, the NRC staff finds the licensee's external events PRA provides an order of magnitude estimate of the external events risk and is therefore acceptable for the ILRT extension application.

Following a peer review of the licensee's internal events PRA against Revision 1 of RG 1.200 and the ASME PRA standard, the licensee performed a gap assessment of its internal events PRA against the current ASME PRA standard and Revision 2 of RG 1.200 and adequately evaluated the findings for applicability to the ILRT extension. The licensee also provided a quantitative assessment of the contribution of external events, including the effects of internal fires, seismic events, high winds, and external flooding using external event PRA models that were previously submitted and reviewed by the NRC staff. Therefore, the NRC staff concludes that the PRA model used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequencies. Accordingly, the first condition is met.

4.3.4 Estimated Risk Increase

The second condition stipulates that the licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small, and consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.5 of the NRC SER for NEI 94-01, Revision 2, and 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 core damage frequency and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. As stated in Attachment 1 of STP's LAR, STP Units 1 and 2 do not rely on containment over-pressure for ECCS performance. Thus, the associated risk metrics include LERF, population dose, and CCFP.

The licensee reported the results of the plant-specific risk assessment in Section 4.11.3 of Attachment 1 to the LAR. Details of the risk assessment are provided in Attachment 4 to the LAR. The reported risk impacts are based on a change in test frequency from three tests in 10 years (the test frequency under 10 CFR 50 Appendix J, Option A) to one test in 15 years. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A ILRT frequency:

1. The reported increase in LERF is 5.27E-08/year for both units, which includes both internal and external events, and the potential effects of liner corrosion. The risk contribution from external events includes the effects of internal fires, seismic events, high winds, and external flooding, as discussed in Section 4.3.3 of this SE. This change in risk is considered to be "very small" (i.e., below 1E-07/year) per the acceptance guidelines in RG 1.174 and is therefore acceptable.

2. The reported change in Type A ILRT frequency from three in 10 years to once in 15 years results in a reported increase in the total population dose of 0.123 person-rem/year for both units. The reported increase in total population dose is below the values provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2. Thus, this increase in the total population dose for the proposed change is considered small and supportive of the proposed change.

3. The increase in CCFP due to change in test frequency from three in 10 years to once in 15 years is 0.87 percent for both units. This value is below the acceptance guidelines in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2.

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

4.3.5 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 the table in Section 4.11.1 of Attachment 1 to the LAR, the methodology in EPRI TR-1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the STP's plant-specific risk assessment. Accordingly, the third condition is met.

4.3.6 Applicability if Containment Overpressure 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 4.11.1 of Attachment 1 to the LAR, the licensee stated that containment over-pressure is not relied upon for ECCS performance for STP. Accordingly, the fourth condition is met.

4.3.7 Conclusion

The licensee performed a plant-specific risk impact assessment for extending the Type A containment ILRT interval to 15 years in accordance with EPRI TR-1009325, Revision 2. As part of this assessment, the licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2, which are listed in Section 4.2 of the NRC SER for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2. Specifically, the NRC staff concludes that the

PRA technical adequacy and estimated risk increases are acceptable for this application, given that the licensee assumed the appropriate leak rate for accident case 3b, and STP Units 1 and 2 do not rely on containment over-pressure. Based on the above, the NRC staff concludes that the proposed permanent extension of the Type A containment ILRT frequency to once in 15 years for STP Units 1 and 2 is acceptable.

5.0 STATE CONSULTATION

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

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 and changes surveillances. 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 published in the *Federal Register*. 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.

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Dan Hoang, NRR/DE/EMCB

Date: April 29, 2016

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA JKlos for/

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 210 to NPF-76
2. Amendment No. 197 to NPF-80
3. Safety Evaluation

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DATE	4/29/16	4/29/16	4/29/16

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