Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

April 29, 2015
NOC-AE-15003227
10 CFR 50.90
File No. G25

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
South Texas Project (STP), Units 1 and 2 License Amendment Request for
Extending the 10 year ILRT to 15 years

In accordance with the provisions of 10 CFR 50.90, STP Nuclear Operating Company (STPNOC) hereby requests a license amendment to South Texas Project Operating Licenses NPF-76 and NPF-80. This proposed license amendment revises Administrative Controls Technical Specification (TS) 6.8.3.j, "Containment Leakage Rate Testing Program".

This License Amendment Request (LAR) and respective TS change reflects a change to extend the Integrated Leak Rate Test (ILRT) performance interval from 10 years to every 15 years in accordance with NEI 94-01, Revision 2A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J".

This letter contains four attachments (Attachments 1 through 4) all of which are non-proprietary documents. Attachment 1 provides an evaluation of the proposed change, a determination that the proposed amendment contains No Significant Hazards Consideration, and the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement pursuant to 10 CFR 51.22(c)(9). The proposed TS marked-up page is included as Attachment 2, and a retyped proposed Technical Specification page is included in Attachment 3. Attachment 4 is probability risk assessment for the permanent extension for the ILRT.

The STPNOC Plant Operations Review Committee has reviewed and concurred with the proposed change to the Technical Specifications.

STPNOC requests approval of this license amendment application by April 30, 2016. This license amendment is scheduled to support the Fall of 2016 Unit 2 refueling outage 2RE18. The requested review period is consistent with NRC internal guidance. STPNOC requests a 90-day implementation period after the amendment is approved.

STI: 34060265

A017
NRR

In accordance with 10 CFR 50.91(b), STPNOC is notifying the State of Texas of this request for license amendment by providing a copy of this letter and attachments.

There are no commitments in this letter.

If there are any questions regarding the proposed amendment, please contact Rafael Gonzales at (361) 972-4779 or me at 361-972-7566.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on April 29, 2015
Date

G.T. Powell
Site Vice President

rjg

Attachments:

1) Proposed Amendment to Technical Specification 6.8.3.j for a Permanent Change in 10CFR50 Appendix J Integrated Leakage Rate Test Interval
2) Markup of Technical Specification Page 6.8.3.j
3) Typed Technical Specification Page 6.8.3.j
4) PRA Evaluation Permanent ILRT Extension Risk Assessment

cc:

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ATTACHMENT 1

PROPOSED AMEMDMENT TO TECHNICAL SPECIFICATION 6.8.3.j
FOR A PERMANENT CHANGE IN 10CFR50 APPENDIX J
INTEGRATED LEAKAGE RATE TEST INTERVAL

LICENSEE'S EVALUATION

1.0 PROPOSED CHANGE

Pursuant to 10CFR50.90, STP Nuclear Operating Company (STPNOC) requests an amendment to the South Texas Project (STP) Unit 1 Operating License (NPF-76) and Unit 2 Operating License (NPF- 80) by incorporating the attached change into the STP Unit 1 and 2 Technical Specifications.

The proposed change to the Technical Specifications (TS) would revise STP TS 6.8.3.j, by replacing the reference to Regulatory Guide (RG) 1.163 (Reference 1) with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01 Revision 2-A (Reference 3), dated October 2008, as the implementation documents used by STPNOC to implement the Unit 1 and Unit 2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

The proposed change would allow an increase in the Integrated Leak Rate Test (ILRT) test interval from its current 10-year frequency to a maximum of 15 years. The frequency of local leakage rate testing of containment isolation valves (Type C) and pressure-retaining or leakage-limiting boundaries other than valves (Type B) are not affected by the adoption of NEI 94-01 Revision 2-A. The proposed change would also delete the listing of one-time exceptions in TS 6.8.3.j previously granted to Integrated Leak Rate Test (ILRT) test frequencies.

STP Technical Specification 6.8.3.j, "Containment Leakage Rate Testing Program" currently states, in part:

> 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 Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program", dated September 1995. The current ten-year interval between performance of the integrated leakage 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 change to STP Technical Specification 6.8.3.j, "Containment Leakage Rate Testing Program" will replace the reference to Regulatory Guide 1.163 with a reference to topical report NEI 94-01 Revision 2-A and delete the last sentence of the paragraph. The proposed change will revise Technical Specification 6.8.3.j to 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 2-A, dated October 2008.

This proposed change is requested to extend the performance of the next Unit 2 ILRT from the Fall 2016 refueling outage to a subsequent refueling outage no later than Spring 2021. This proposed amendment would also extend the performance of the next Unit 1 ILRT to be performed no later than Fall 2024.

Attachment 4 contains the plant specific risk assessment conducted to support this proposed change. This risk assessment followed the guidelines of Nuclear Regulatory Commission (NRC) Regulatory Guide 1.174 (Reference 4) and NRC Regulatory Guide 1.200, Revision 2 (Reference 5). The risk assessment concluded that the increase in risk as a result of this proposed change is small and is well within established guidelines.

2.0 BACKGROUND

2.1 Justification for the Technical Specification Change

2.1.1 Chronology of Testing Requirements of 10 CFR 50 Appendix J

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. 10 CFR 50, Appendix J also ensures that periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the systems and components penetrating primary containment. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident. Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the primary containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for primary containment penetrations; and (3) Type C tests, intended to measure containment isolation valve leakage rates. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

In 1995, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term "performance-based" in 10 CFR 50 Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

Also in 1995, Regulatory Guide (RG) 1.163 was issued. The RG endorsed Nuclear Energy Institute (NEI) 94-01, Revision 0, (Reference 6) with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency for the containment Type A (ILRT) test from three tests in 10 years to one test in 10 years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, (Reference 7) and Electric Power Research Institute (EPRI) TR-104285 (Reference 8) both of which showed that the risk increase associated with extending the ILRT surveillance interval was very small. In addition to the 10-year ILRT interval, provisions for extending the test interval an additional 15 months were considered in the establishment of the intervals allowed by RG 1.163 and NEI 94-01, but that this "should be used only in cases where refueling schedules have been changed to accommodate other factors."

In 2008, NEI 94-01, Revision 2-A, (Reference 3) was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, subject to the limitations and conditions noted in Section 4.0 of the NRC Safety Evaluation Report (SER) on NEI 94-01. The NRC SER was included in the front matter of this NEI report. NEI 94-01, Revision 2-A, includes provisions for extending Type A ILRT intervals to up to fifteen years and incorporates the regulatory positions stated in Regulatory Guide 1.163 (September 1995). It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights.

The NRC has provided the following concerning the use of grace in the deferral of ILRTs past the 15 year interval in NEI 94-01, Revision 2-A, NRC SER Section 3.1.1.2:

> "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."

In 2012, NEI 94-01, Revision 3-A (Reference 2), was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J and includes provisions for extending Type A ILRT intervals up to fifteen years. NEI 94-01 has been endorsed by Regulatory Guide 1.163 and NRC SERs of June 25, 2008 (Reference 9) and June 8, 2012 (Reference 10) as an acceptable methodology for complying with the provisions of Option B to 10 CFR Part 50. The regulatory positions stated in Regulatory Guide 1.163 as modified by NRC SERs of June 25, 2008 and June 8, 2012 are incorporated in this document. The NRC SER was included in the front matter of this NEI report. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights. Extensions of Type B and 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. Intervals may be increased from 30 months up to a maximum of 120 months for Type B tests (except for containment airlocks) and up to a maximum of 75 months for Type C tests. If a licensee considers extended test intervals of greater than 60 months for Type B or Type C tested components, the review should include the additional considerations of as-found tests, schedule and review as described in NEI 94-01, Revision 3-A, Section 11.3.2.

STPNOC has evaluated the extended Type C intervals afforded by NEI 94-01 Revision 3-A and based on the 18 month operating cycles of STP Units 1 and 2 there is no benefit, i.e., further extension of Type C intervals, to be derived by adopting NEI 94-01 Revision 3-A.

2.1.2 Current STPNOC 10 CFR 50 Appendix J Requirements

Title 10 CFR Part 50, Appendix J was revised, effective October 26, 1995, to allow licenses to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance Based Requirements." STPNOC has implemented the requirements of 10 CFR Part 50, Appendix J, Option B for Type A, B and C tests. Current Technical Specification 6.8.3.j requires that a program be established to comply with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR Part 50 Appendix J, Option B, as modified by approved exemptions. The program is required to be in accordance with the guidelines contained in Regulatory Guide 1.163. Regulatory Guide 1.163 endorses, with certain exceptions, NEI 94-01 Revision 0 as an acceptable method for complying with the provisions of Appendix J, Option B.

Regulatory Guide 1.163, Section C.1 states that licensees intending to comply with 10 CFR Part 50, Appendix J, Option B, should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 (Reference 6) rather than using test intervals specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.8-1994. NEI 94-01 Revision 0, Section 11.0 refers to

Section 9.0, which states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per ten years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage was less than 1.0 L_a (where L_a is the maximum allowable leakage rate at design pressure). Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Adoption of the Option B performance based containment leakage rate testing program altered the frequency of measuring primary containment leakage in Types A, B, and C tests but did not alter the basic method by which Appendix J leakage testing is performed. The test frequency is based on an evaluation of the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded. The allowed frequency for Type A testing as documented in NEI 94-01 Revision 0, is based, in part, upon a generic evaluation documented in NUREG-1493 (Reference 7). The evaluation documented in NUREG-1493 included a study of the dependence or reactor accident risks on containment leak tightness for differing types of containment types, including a reinforced, shallow domed concrete containment similar to STP containment structures. NUREG-1493 concluded in Section 10.1.2 that reducing the frequency of Type A tests (ILRT) from the original three tests per ten years to one test per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Types B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements. Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, NUREG-1493 concluded that increasing the interval between ILRTs is possible with minimal impact on public risk.

2.1.3 STPNOC 10 CFR 50 Appendix J Option B Licensing History

SER dated May 19, 1995 - ML021300263 (Reference 11)
By application dated March 16, 1995, Houston Lighting & Power Company, et al., (the licensee) requested changes to the STP Facility Operating License Nos. NPF-76 and NPF-80) for the South Texas Project - Units 1 and 2. The proposed changes revised TS 4.6.1.2, regarding the test frequency requirements for the overall integrated containment leakage rate tests, so that it would reference 10 CFR Part 50, Appendix J and approved exemptions, rather than paraphrase the regulation.

SER dated September 7, 1995 - ML021330525 (Reference 12)
The NRC issued Amendment Nos. 80 and 69 to Facility Operating License Nos. NPF-76 and NPF-80 for the South Texas Project, Units 1 and 2.

The amendments revised the TSs on containment leakage, making the action statement consistent with the need to perform Type C testing at power, and replacing the surveillance requirements with a single requirement to apply the requirements of Appendix J as modified by approved exemptions. The amendments also revised the TSs on containment integrity, containment leakage, and containment air locks, to eliminate the numerical value of calculated peak containment internal pressure related to the design basis accident.

SER dated August 13, 1996 - ML021300572 (Reference 13).
South Texas Project, Units 1 and 2 -Amendment Nos. 84 and 71 to Facility Operating License Nos. NPF-76 and NPF-80 (TAC Nos. M94536 and M94538).

The amendments changed the TSs to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." Part of the requested change, regarding the frequency of leakage rate testing the normal containment purge valves and the supplementary containment purge valves, was not granted. The 1996 test intervals were not based on Appendix J considerations and the proposed License Amendment Request was outside the scope of the proposed change to Option B.

SER dated August 3, 2001 - ML011990368 (Reference 14)
South Texas Project, Units 1 and 2 - Safety Evaluation on Exemption Requests From Special Treatment Requirements of 10 CFR Parts 21, 50, and 100 (TAC NOS. MA6057 and MA6058). Exemptions from certain requirements of 10 CFR Parts 21, 50, and 100.

SER dated September 17, 2002 - ML022410163 (Reference 31)
The Commission has issued the enclosed Amendment No. 143 to Facility Operating License No. NPF-76 and Amendment No. 131 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The change revises the first paragraph of TS Subsection 6.8.3.j to read: "The current ten-year interval between performance of the integrated leakage 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)."

SER dated January 7, 2003 - ML030130435 (Reference 32)
The Commission has issued the enclosed Amendment No. 147 to Facility Operating License No. NPF-76 and Amendment No. 135 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively.

The amendments revised TS 3/4.6.1.7, "Containment Ventilation System", to extend the intervals between operability tests of the normal and supplementary containment purge valves, from 6 and 3 months, respectively, to 18 months for both.

2.2 Containment Building Description

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 unbonded wire tendons.

A continuous, reinforced-concrete tendon gallery is located at the perimeter of the mat with floor of the gallery extending 5'-6" below the base of the mat. The gallery is about 8 ft wide (7'-8" in Unit 1 and 8'-0" in Unit 2) and 11 ft high and is provided for the installation and surveillance of the vertical post-tensioning system. The bottom of the tendon gallery is 67'-3" below grade. Access to the tendon gallery is provided by a shaft from the ground level to the tendon gallery. Emergency access to the gallery is provided through the Mechanical-Electrical Auxiliaries Building.

The Containment wall is independent of the adjacent interior and exterior structures, with sufficient space being provided between the Containment wall and the adjacent structures to prevent contact under all loading conditions.

2.2.1 Dimensions of Containment:

- Inside diameter: 150 ft.
- Inside height to top of the dome: 239-1/4 ft.
- Thickness of cylindrical walls: 4 ft.
- Thickness of dome: 3 ft.
- Foundation mat thickness: 18 ft.
- Top of the foundation mat: approximately 41-1/4 ft. below grade
- Containment design pressure: 56.5 psig
- Containment design temperature: 286 °F

2.2.2 Steel Liner

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.

An anchorage system is provided to prevent instability of the liner. For the dome, the anchorage system consists of meridional structural tees, circumferential angles, and plates, while for the cylinder, a system of vertical and circumferential stiffeners, using structural angles, channels, and plates, is provided.

Leak chase channels and angles are provided at the bottom liner seams, which, after construction, are inaccessible for leaktightness examination due to the 2-ft interior, fill slab.

2.2.3 Post-Tensioning Tendons

The cylindrical portion and the hemispherical dome of the Containment are prestressed by a post-tensioning system consisting of horizontal and vertical tendons. Three buttresses are equally spaced at 120 degrees around the Containment.

The cylinder and the lower half of the dome are prestressed by horizontal tendons anchored 360 degrees apart, bypassing the intermediate buttresses. Each successive hoop tendon is progressively offset 120 degrees from the one beneath it. The vertical U-shaped tendons are continuous over the dome, forming a two-way system for the dome. These tendons are anchored in the continuous gallery beneath the base mat.

The tendons are placed in embedded-tendon sheaths, which are filled with a corrosion inhibitor.

2.2.4 Containment Penetrations and Attachments

Access into the Reactor Containment Building (RCB) is provided by an equipment hatch, a personnel airlock, and an auxiliary airlock. The equipment hatch is a 24-foot inside diameter, single-closure penetration. It consists of a welded steel barrel furnished with a double O-ring gasket and a bolted, dished door. The personnel airlock is an 11-foot-6-inch inside diameter, welded-steel assembly with double doors. The auxiliary airlock is a 5-foot-5-inch inside diameter, welded-steel assembly with double doors.

Other penetrations through the Containment include the electrical penetrations, the piping penetrations, and the fuel transfer tube. All penetrations are pressure-resistant, leaktight, welded assemblies. The penetration sleeves are welded to the liner and anchored into the concrete Containment wall.

The fuel transfer tube penetration between the refueling canal in the RCB and the spent fuel pool in the Fuel Handling Building (FHB) consists of a stainless steel pipe inside a carbon steel sleeve. The inner pipe acts as a transfer tube; the outer tube is welded to the Containment liner. Bellows expansion joints are provided to permit differential movement.

Canister-type penetrations are used for electrical conductors passing through the Containment. The penetration canisters are installed in steel penetration sleeves welded into the wall of the Containment liner. Sealing between the canisters and the sleeves is accomplished by welding.

Piping penetration assemblies are generally of three types, the type of penetration used for a particular line being dependent on the service requirements of that line. A high-energy penetration is used where the temperature or pressure of the fluid is high and considerable thermal movement of the line can be expected. Moderate-energy penetrations are used where little or no thermal movement of the process line is anticipated. Multiple penetrations are used where more than one pipe goes through a penetration.

The crane girder support brackets are welded to a section of the liner plate and anchored into the Containment concrete wall.

3.0 REGULATORY ANALYSIS

3.1 Applicability

The testing requirements of 10CFR50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified In the Technical Specifications. Limiting containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident.

10CFR50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." STPNOC previously selected Option B. Regulatory Guide 1.163, dated September 1995, specified a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8 - 1994 (Reference 1), subject to several regulatory positions in the guide.

Exceptions to the requirements of Regulatory Guide 1.163 are allowed by 10CFR50, Appendix J, Option B, Section V.B, Implementation," which states:

> The Regulatory Guide or other implementing document used by a licensee, or applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

NEI 94-01 Revision 2-A, dated October 2008 was approved for use by the NRC through the issuance of the following:

> 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" (TAC NO. MC9663) dated June 25, 2008.

The NRC staff reviewed NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2. For NEI TR 94-01, Revision 2, the NRC staff determined that it describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J. This guidance includes provisions for extending Type A ILRT intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. The NRC staff finds that the Type A testing methodology as described in ANSI/ANS-56.8-2002, and the modified testing frequencies recommended by NEI Topical Report (TR) 94-01, Revision 2, serves to ensure continued leakage integrity of the containment structure. Type B and Type C testing ensures that individual penetrations are essentially leak tight. In addition, aggregate Type Band Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

For EPRI Report No. 1009325, Revision 2, a risk-informed methodology using plant-specific risk insights and industry ILRT performance data to revise ILRT surveillance frequencies, the NRC staff finds that the proposed methodology satisfies the key principles of risk-informed decision making applied to changes to TSs as delineated in RG 1.177 and RG 1.174.

The NRC staff, therefore, found that this guidance 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 this SE.

Therefore, the change proposed by this application does not require an exemption from 10 CFR 50 Appendix J, Option B.

3.2 Test Frequency

The surveillance frequency for Type A testing in NEI 94-01 Revision 2-A is at least once per 15 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 La) and consideration of the performance factors in NEI 94-01, Section 11.3. Adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B, and C tests.

Frequency is based upon an evaluation which looks at the as found leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493 (Reference 7). NUREG-1493 made the following observations with regard to decreasing the test frequency:

- Reducing the Type A (ILRT) testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because an ILRT will identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks found by Type A tests have been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the same fraction of leakage detected solely by Type A testing, the interval between integrated leakage rate testing can be increased with minimal effect on public risk.

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Because leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

3.3 Continuation of Type B and C Tests

The existing Appendix J Type B and Type C testing programs will not be modified by this proposed change. However, on August 3, 2001, the South Texas Project received an exemption of low safety/risk significant (LSS) and non-risk significant (NRS) components from the special treatment of 10 CFR 50 requirements, including Appendix J Type C testing (Reference 34).

The staff found that the licensee's application of a risk-informed categorization process has identified a class of SSCs that have little or no safety significance with respect to protecting the health and safety of the public. The staff also found that the proposed treatment processes to be applied to activities associated with LSS and NRS SSCs, as described by the licensee, if effectively implemented, will provide reasonable confidence that safety-related LSS and NRS SSCs remain capable of performing their safety functions under design-basis conditions. Further, the staff found that leakage through containment isolation valves meeting the licensee's criteria would have negligible impact on public health and safety. As such the staff found that it was reasonable for STPNOC to assert that the cumulative limits for containment leakage would be based upon the tested components, with the assumption that the exempted components contribute zero leakage. Based on this finding, the staff determined that an exemption from the 10 CFR Part 50, Appendix J, Option B, Section III.B requirement that "the sum of the leakage rates at accident pressure of Type B tests and pathway leakage rates from Type C tests; must be less than the performance criterion (La) with margin, as specified in the Technical Specifications," is not necessary.

Based on these findings, the staff concluded that granting of the requested exemption from the Type C testing requirements of 10 CFR Part 50, Appendix J, Option B, Section III.B, for LSS and NRS containment isolation valves that meet the licensee's proposed criteria discussed and evaluated above, would pose no undue risk to public health and safety. The staff found that the categorization process was not considered when the requirements of 10 CFR Part 50, Appendix J, Option B, Section III.B, were adopted and that it is in the public interest to grant an exemption from the special treatment requirements. This satisfies the special circumstance of 10 CFR 50.12(a)(2)(vi). Therefore, the staff determined that the exemptions should be granted from the Type C testing requirements of 10 CFR Part 50, Appendix J, Option B, Section III.B, as requested by STPNOC.

4.0 TECHNICAL ANALYSIS

4.1 Inspections

4.1.1 Service Level 1 Coating Assessment inside Reactor Containment

The Coating Assessment of Service Level 1 coatings in containment is a visual examination of all accessible concrete and steel coated surfaces to identify any type of coating degradation such as peeling, flaking, blistering, delamination, rusting and mechanical damage. Any areas of coating degradation are documented on the Coating Walkdown Checklist and evaluated for severity and determined to be repaired during the current outage, repaired in the next available outage or continued to be monitored and re-evaluated during the next available outage.

The Coating Assessment includes a visual examination of all accessible Service Level 1 coatings inside containment including the steel containment liner, structural steel, supports, penetrations, uninsulated equipment, and concrete walls and floors receiving epoxy surface systems. This includes areas near sumps associated with the emergency core cooling system. The Coating Assessment does not include coating of surfaces that are insulated or otherwise enclosed in normal service and concrete receiving a non-film forming clear sealer coat only.

The Coating Assessment of Service Level 1 coatings are conducted at each refueling outage by individuals meeting the educational, professional achievements, and nuclear coatings experience qualification criteria as a Nuclear Coatings Specialist in accordance with ASTM D7108-05. The Coating Engineer is the responsible engineer in charge of the safety-related coatings program. The Coating Planner is a certified NACE Inspector and meets the educational, professional achievements and nuclear coatings experience qualification criteria for qualification as a Nuclear Coatings Specialist in accordance with ASTM D7108-05.

Prior to the Coating Assessment, the Nuclear Coatings Specialist reviews the two previous coating assessment reports. From the previous two coating assessment reports, areas identified as being monitored and re-evaluated in the next available outage are noted and added to the location maps for the current Coatings Assessment as applicable.

The coating assessments are conducted using location maps dividing the RCB into twenty-four (24) identifiable floor plans labeled with pertinent elevation, azimuth references, structural features and components. All areas of degraded coatings identified during Coatings Assessments are recorded on the location maps. All areas that cannot be assessed during the Coating Assessment and the specific reason why the inspection cannot be conducted are identified on the location maps as applicable.

Physical test are performed on an as-need basis as determined by the Nuclear Coatings Specialist. Blistering of all sizes as well as Flaking, Peeling and Delamination are considered rejectable conditions. The source and extent of rusting is evaluated during the visual assessment by the Nuclear Coatings Specialist.

The Coating Assessment Reports are evaluated and approved by the Nuclear Coatings Specialist who collaborate in the evaluation of degraded coatings and determination of recommendations. The Coating Engineer prepares the Coatings Assessment Report. Work Orders are prepared for degraded coatings in accordance with Condition Reporting Process.

4.1.2 Results of Recent Coatings Inspections

1RE18 Service Level 1 Coating Assessment Report dated 4/17/14 characterized the Service Level 1 coatings inside the Unit 1 Reactor Containment Building as being in very [good] condition. The coating assessment performed by qualified personnel did not identify any areas of blistering, peeling, flaking or delamination. The identified areas of coating degradation include minor surface rusting of structural steel, hangers, pipe supports, and pipe. All coating degradations were entered into the Condition Reporting Process and will be repaired during the next available outage. Bare metal bolts were also captured in this assessment and entered into the Condition Reporting Process. The identified bare metal bolts will be coated during the next available outage. Areas of identified liner degradation are identified as follows:

Report No. 1RE18-0011
Location: 68', Az. 270°, Room 501
Identified Degradation: In May 2000 an area estimated to be 4" in height and 6 to 8 inches in length, located at azimuth 271° near elevation 90 ft, was identified as an indication on the containment wall by the reactor vessel head lift rig. Engineering investigated to determine the condition of the indication under CR 00-8532-1 and determined that the outer coating was removed with the primed surface below exposed with no signs of corrosion or coating deterioration noted; the condition was found acceptable as-is.
Determination: Monitor and re-evaluate in next available outage.
CR No.: N/A
Work Order No.: N/A
(1RE18 Note: The indication was re-evaluated in 1RE16, 1RE17, and 1RE18 and found to be approximately the same size, dark in color (primer coat) and showing no signs of corrosion within the area and no streaks of rust on the wall below the area. The size of the indication or its condition has not changed since May 2000)

Report No. 1RE17-021
Location: 19', Az. 352°, Room 210
Identified Degradation: Minor surface rust on liner wall approximately 2 inches in diameter. Area appears to be a previous repair area, WO to include scaffold for access and hold point for engineering to assess.
Determination: Repair in next available outage
CR No.: 13-1211-2
Work Order No.: 540960
(1RE18 Note: Reworked in 1RE18 with WO 540960 closed 3/30/14)

2RE16 Service Level 1 Coating Assessment Report dated 12/2/13 characterized the Service Level 1 coatings inside the Unit 2 Reactor Containment Building as being in very [good] condition. The coating assessment performed by qualified personnel did not identify any areas of blistering, peeling, flaking or delamination. The identified areas of coating degradation include minor surface rusting on three pipe supports. All coating degradations were entered into the Condition Reporting Process and will be repaired during the next available outage or monitored and re-evaluated in the next available outage. Areas of identified liner degradation are identified as follows:

The 2RE15 assessment identified mechanical damage to liner coatings at EL-2, AZ 278°. The area was reinspected on 2/1/2012 and it was determined that the damaged coatings does not affect the function of the liner and was likely caused by a scaffold pole bumping into the liner plate. This is a cosmetic rework.
(Scheduled for rework in 2RE16, CR 12-10788-11, WO 528798)
(Reworked in 2RE16 with WO 529798 closed 11/20/13)

4.1.3 Inservice Inspection Program For Concrete Containment - IWL

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 No Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1 in accordance with 10CFR50.55a(b)(2)(vi). The additional requirements specified by 10CFR50.55a(b)(2)(viii) are identified in the program procedure. The ISI Program Plan for the THIRD interval was developed in accordance with the requirements of 10CFR50.55a that became effective on October 15, 2008.

This program complies with the requirements of ASME Section XI Code, Table IWL-2500-1, Category L-A & L-B.

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.

The Concrete Containment inservice inspections shall be implemented on a 5-year inspection interval [IWL-2410(a)]. The start and end dates for Interval 3 and 4 of the Concrete Containment inservice inspection activities are shown in Table 4.1.3-1.

Table 4.1.3-1, 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

The surveillance associated with the Concrete Containment is implemented through specification 4C23HCS0001, Inservice Surveillance of Containment Post-Tensioning System.

The ASME Class CC (IWL) equivalent accessible components subject to examination are:

- Concrete surfaces, and;

- Unbonded Post-Tensioning systems (Tendon, Wire or strand, Anchorage hardware and surrounding concrete, Corrosion protection medium, and any free water in ducts).

The rules of IWL-1220 have been used to EXEMPT components from examination. Component's exempt from examination are listed below:

- Portions of the concrete surface that are covered by the liner, foundation material, or back fill, or are otherwise obstructed by adjacent structures, components, parts, or appurtenances.

All tendon anchorages are accessible. However, some (above the power operated relief valves) have been deemed unsafe to access at power. These were NOT exempted by IWL 2521.1, but were instead exempted by relief request approved by the NRC.

Personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification requirements of IWA-2300 of the 2004 Edition No Addenda of ASME Section XI. [50.55a(b)(2)(viii)(F)]

Inaccessible areas
When conditions exist in accessible areas that could indicate the presence of or result in degradation to an inaccessible area, then 10CFR50.55a(b)(2)(viii)(E) requires an evaluation be performed to determine the acceptability of the inaccessible area. A Condition Report (CR) per 0PGP03-ZX-0002 (Condition Reporting Process) shall be generated to document the evaluation. The inaccessible area evaluation shall include the following information. The information shall be provided in the Inservice Inspection Summary Report during the outage preceding completion of the inspection activities.

- A description of the type and estimated extent of degradation, and the conditions that led to the degradation;

- An evaluation of each area, and the result of the evaluation, and;

- A description of necessary corrective actions

STPNOC has not needed to implement any new technologies to perform inspections of any inaccessible areas at this time. However, STPNOC actively participates in various nuclear utility owners groups and ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its applicability to STP. Adjustments to inspection plans and availability of new, commercially available technologies for the examination of the inaccessible areas of the containment would be explored and considered as part of these activities.

4.1.4 Inservice Inspection Program for Containment Metal Liner - IWE

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 No Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1 in accordance with 10CFR50.55a(b)(2). The additional requirements specified by 10CFR50.55a(b)(2)(ix) are identified in the program procedure.

This program complies with the requirements of Code Table IWE-2500-1, Examination Category E-A & E-C.

The schedules for the 2nd and 3rd IWE Intervals are shown in Tables 4.1.4-1 and 4.1.4-2 below.

Table 4.1.4-1, STP Unit 1, IWE Containment Metal Liner Examination Schedule

Interval	Period	Dates	Outage	Dates
2	1	9/9/09 – 9/8/13	1RE16	4/3 – 4/27/11
			1RE17	10/20 – 11/19/12
	2	9/9/13 – 9/8/16	1RE18	3/29 - 4/20/14
			1RE19	10/3 - 11/12/15
	3	9/9/16 – 9/8/19	1RE20	4/1 - 4/23/17
			1RE21	10/7 – 11/4/18
3	1	9/9/19 – 9/8/23	1RE22	3/2020*
			1RE23	10/2021*
	2	9/9/23 – 9/8/26	1RE24	3/2023*
			1RE25	10/2024*
	3	9/9/26 – 9/8/29	1RE26	3/2026*
			1RE27	10/2027*

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

Table 4.1.4-2, STP Unit 2, IWE Containment Metal Liner Examination Schedule

Interval	Period	Dates	Outage	Dates
2	1	9/9/09 – 9/8/13	2RE14	3/27 – 5/2/10
			2RE15	10/29 - 11/22/11
			2RE16	4/27 – 5/24/13
	2	9/9/13 – 9/8/16	2RE17	10/4 – 10/29/14 –

Interval	Period	Dates	Outage	Dates
			2RE18	4/2 – 4/28/16
	3	9/9/16 – 9/8/19	2RE19	9/30 – 10/24/17
			2RE20	3/31 – 4/22/20
3	1	9/9/19 – 9/8/23	2RE21	10/2021*
			2RE22	3/2023*
			2RE23	10/2024*
	2	9/9/23 – 9/8/26	2RE24	3/2026*
			2RE25	10/2027*
	3	9/9/26 – 9/8/29	2RE26	3/2029*
			2RE27	10/2030*

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

The ASME Class MC (IWE) equivalent accessible components subject to examination are:

- Shell and dome containment metallic liner pressure boundary plate, including reinforcing plates around penetrations and openings.
- Structural stiffeners, including attachment welds. Attachment welds between structural attachments and the containment liner pressure-retaining boundary.
- Metal liner anchorage.
- Fuel transfer tube.
- Embedded plates which are part of the containment pressure retaining metallic liner, such as Polar Crane Girder Support Bracket embedded plates.
- Penetration sleeves, penetration metallic liners, bellows and reinforcement and any attached pressure-retaining connections such as pipe caps or flanges.
- Equipment Hatch, Personnel Air Lock, and Auxiliary Personnel Air Lock, including any structural reinforcements.
- Blind flanges or bolted covers attached to containment penetration flanges, including electrical penetration flanges.
- Bolting connecting pressure retaining parts to the containment, including Equipment Hatch latch bolts, bolting on spare penetration covers, etc.

- Moisture barriers (including caulking, flashing, sealants or other devices used to prevent intrusion of moisture against the pressure retaining metal containment shell or liner) at containment internal concrete-to-metal interface.

The rules of Code article IWE-1220 have been used to EXEMPT components from examination. Component's EXEMPT from examination are listed below:

- Vessels, parts, and appurtenances that are outside the boundaries of the containment, as defined in the Design Specifications;

- Embedded or inaccessible portions of containment vessels, parts, and appurtenances that met the requirements of the original Construction Code;

- Portions of containment vessels, parts, and appurtenances that become embedded or inaccessible as a result of vessel repair/replacement activities if the conditions of Code articles IWE-1232(a) and (b) and IWE-5220 are met;

- Piping, pumps, and valves that are part of the containment system, or which penetrate or are attached to the containment vessel. These components SHALL be examined in accordance with the rules of Code articles IWB or IWC, as appropriate to the classification defined by the Design Specifications.

The requirements of Code article IWE-1231 SHALL be met to maintain accessibility for Containment Metal Liner components for the life of the plant. Eighty (80) percent of the pressure –retaining boundary (excluding attachments, structural reinforcement, and areas made inaccessible during construction), SHALL remain accessible. CISI Figure drawings document that 84% is accessible. Inaccessible surface areas are defined in Code article IWE-1232.

Moisture Barriers – General Visual (E-A, Item E1.30)

The moisture barrier at STP is the concrete coating that overlaps onto the liner plate coating between the concrete to metal containment interface at the (-)11 foot elevation.

The examination boundary includes the accessible surface of the sealant (coating) used to prevent moisture intrusion against the pressure retaining metal containment liner at concrete-to-metal interfaces and at metal-to-metal interfaces which are not seal welded. (Table IWE-2500-1, Category E-A, Note 3)

Containment Surfaces Requiring Augmented Examination - Examination Category E-C, Visible Surfaces – Detailed Visual (E-C, Item E4.11) VT-1 [per 50.55a(b)(2)(ix)(G)]

There are currently NO augmented areas identified for Unit 1 or Unit 2 in accordance with Code articles IWE-1240 and IWE-2500.

Surface Area Grid – Minimum Wall Thickness Location – Ultrasonic thickness (E-C, Item E4.12)

There are currently NO augmented areas identified for Unit 1 or Unit 2 in accordance with Code articles IWE-1240 and IWE-2500.

Inaccessible areas
When conditions exist in accessible areas that could indicate the presence of or result in degradation to an inaccessible area, then 10CFR50.55a(b)(2)(ix)(A) requires an evaluation be performed to determine the acceptability of the inaccessible area.

A Condition Report (CR) per 0PGP03-ZX-0002 (Condition Reporting Process) SHALL be generated to document the evaluation.

The inaccessible area evaluation SHALL include the following information. The information SHALL be provided in the Inservice Inspection Summary Report (0PGP04-ZE-0304).

- A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
- An evaluation of each area, and the result of the evaluation, and;
- A description of necessary corrective actions

STPNOC has not needed to implement any new technologies to perform inspections of any inaccessible areas at this time. However, STPNOC actively participates in various nuclear utility owners groups and ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its applicability to STP. Adjustments to inspection plans and availability of new, commercially available technologies for the examination of the inaccessible areas of the containment would be explored and considered as part of these activities.

4.1.5 Results of Recent IWE/IWL Examinations

4.1.5.1 The 25th Year South Texas Project Nuclear Power Plant Units 1 and 2 containment structure post-tensioning system IWL Tendon Surveillance was completed July 2014. A total of eighteen (18) tendons were inspected during the

2014 tendon surveillance of which eleven (11) were taken from Unit 1, and seven (7) from Unit 2. Five (5) of the eleven (11) tendons from Unit 1 were added after issuance of the initial scope. One (1) of the seven (7) tendons from Unit 2 was pre-selected for a visual examination.

Summary of Findings:

- Acceptable corrosion levels (a corrosion level of 1 or 2) were found on all selected tendon ends, and no cracks were found on anchorage components.
- No moisture or any free water was found at either end of any tendon.
- The sheathing filler (grease) samples were sent for chemical analysis to determine levels of chlorides, nitrates and sulfides, as well as moisture content and neutralization numbers. The chemical analysis results indicate all grease samples are within the acceptance criteria.
- After retensioning the hoop tendon 2H-126 of Unit 2, three (3) of the 186 wires were found protruding on the shop end. No other previously unreported missing or protruding buttonheads or wires were discovered on any other tendon end inspected in this surveillance.
- A detailed visual inspection was performed within a 24" perimeter of concrete surrounding the bearing plate of each tendon end inspected. No concrete cracks greater than 0.010 inches wide were found at any of the surveillance tendon ends.
- No tendons were found to have lift-off force values below 95% of their predicted lift-off force.
- The detensioned tendons were retensioned to acceptable force levels, per IWL-2523.
- All elongation measurements were acceptable during retensioning of tendons during this surveillance.
- All test wires removed from detensioned tendons were found to have acceptable corrosion levels, diameter, yield strength, ultimate strength, and elongation
- All inspected tendons were resealed and greased to acceptable levels.
- A comparison of the "As-Found" force levels to the original force levels was made in an effort to detect any evidence of system degradation. Unit 2 maximal force losses since original installation for each tendon group are reported as 17.3% for the hoop tendons, and 14.7% for the vertical

tendons. These values indicate that no abnormal average force differences were observed.

- A general visual of the containment structures did not result in any reportable indications.

The post-tensioning system for South Texas Project Units 1 and 2 continues to meet the design requirements, and no evidence of abnormal degradation was observed or recorded during the 25th Year Tendon Surveillance.

The containment post-tensioning systems are performing in accordance with the design requirements, and are expected to do so for the projected 60-year life of the units.

4.1.5.2 The 25th Year South Texas Project Nuclear Power Plant Units 1 and 2 containment structure IWL Concrete Surveillance was performed in conjunction with the 25th Year South Texas Project Nuclear Power Plant Units 1 and 2 containment structure post-tensioning system IWL Tendon Surveillance completed July 2014.

A general inspection was conducted of the exposed accessible exterior surfaces of the concrete containment. Inspection findings are as follows:

- Containment surface findings included small bug holes, small stress cracks (<0.010" wide), tie grout patches, abandoned ¼" and ½" anchors, as well as painted walls and nails.

- Previously recorded findings included grout patches on construction pour lines cracking and falling out in addition to rust stains from lightening rod anchors.

No Recordable Indications were made during the examination and inspections met acceptance by the STPNOC Responsible Engineer.

4.1.5.3 Containment Liner IWE Inspection

An inspection of the Unit 1 Containment Building liner was conducted during 1RE18, report dated 8/9/2014, and the Unit 2 liner during 2RE15, report dated 2/20/12. In both cases no items were found with flaws or relevant conditions that required evaluation for continued service.

4.2 Operational Containment Venting

During power operation, instrument air leaks from air-operated valves inside containment and pressurizes the containment building. Containment pressure is monitored and conditions approaching the limits allowed by the Technical Specifications are annunciated. The increase in the building internal pressure is reduced by periodic operation of the supplementary purge system. This cycling of the containment pressure during operation amounts to a periodic integrated pressure test of the containment at a low differential pressure. With a large pre-existing leak, operation of the containment purge system would not be necessary, and would be noticed by plant operators.

Although not as significant as pressure resulting from a Design Basis Accident, the fact that the containment can be pressurized by leakage from air-operated valves provides a degree of assurance of containment structural integrity (i.e., no large leak paths in the containment structure). This feature is a complement to visual inspection of the interior and exterior of the containment structure for those areas that may be inaccessible for visual examination.

4.3 Integrated Leakage Rate Testing History (ILRT)

Previous Type A tests confirmed that the STP reactor containment structures have leakage well under acceptance limits and represents minimal risk to increased leakage. Continued Type B and Type C testing for direct communication with containment atmosphere minimize this risk. Also, the Inservice Inspection (IWE/IWL) program and maintenance rule monitoring provide confidence in containment integrity.

To date, four Type A tests, preoperational and operational, have been performed on Unit 1, and three Type A tests, preoperational and operational, have been performed on Unit 2. There is considerable margin between these Type A test results and the Technical Specification 6.8.3.j limit of 0.75 La. where La is equal to 0.3% by weight of the containment air per day at the peak accident pressure. These test results demonstrate that both units have low leakage containments.

Table 4.3-1, Integrated Leakage Rate Testing History

Unit (Date)	Mass Point Leakage (Weight %/Day)	Acceptance Limit (Weight %/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 plants Technical Specifications 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 tests performed previously.

4.4 Containment Leakage Rate Testing Program

4.4.1 Type B and Type C Testing Program

STP Type B and C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and containment isolation valves in accordance with 10 CFR Part 50, Appendix J, Option B, and Regulatory Guide 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Type B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. Per Technical Specification 6.8.3.j, the allowable maximum pathway total Types B and C leakage is 0.6 L_a where 0.6 L_a equals approximately 455,050 sccm.

As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority of all potential containment leakage paths. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

STPNOC received an approved exemption from the NRC for the type C test requirements in Appendix J, Option B, Section III.B to 10 CFR Part 50, to the extent that those requirements pertain to containment isolation valves that meet the following criteria:

1. The valve has been categorized as low safety significant (LSS) or non-risk significant (NRS); and

2. The valve meets one or more of the following criteria:

   a. The valve is required to operate (i.e., open) under accident conditions to prevent or mitigate core damage events (e.g., CC-MOV-0057, Component Cooling Water to Reactor Containment Fan Coolers).

   b. The valve is normally closed and in a physically closed, water-filled system. (e.g., containment isolation valves in the Demineralized Water system)

   c. The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary (e.g., containment isolation valves in the Component Cooling Water system and in the Instrument Air system).

   d. The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating, and is connected to the reactor coolant pressure boundary. The process line between the containment isolation valve and the reactor coolant pressure boundary is nonnuclear safety were it not for the fact that it penetrates the containment building). An example is the Safety Injection accumulator nitrogen supply valve.

e. The valve size is 1 inch NPS or less (i.e., by definition the valve failure does not contribute to large early release).

The NRC staff found that the licensee's application of a risk-informed categorization process had identified a class of SSCs that have little or no safety significance with respect to protecting the health and safety of the public. The staff also found that the proposed treatment processes to be applied to activities associated with LSS and NRS SSCs, as described by the licensee, if effectively implemented, will provide reasonable confidence that safety-related LSS and NRS SSCs remain capable of performing their safety functions under design-basis conditions. Further, the staff found that leakage through containment isolation valves meeting the licensee's criteria would have negligible impact on public health and safety. As such the staff found that it was reasonable for STPNOC to assert that the cumulative limits for containment leakage would be based upon the tested components, with the assumption that the exempted components contribute zero leakage. Based on this finding, the staff determined that an exemption from the 10 CFR Part 50, Appendix J, Option B, Section III.B requirement that the sum of the leakage rates at accident pressure of Type B tests and pathway leakage rates from Type C tests, must be less than the performance criterion (La) with margin, as specified in the Technical Specifications, is not necessary.

Based on these findings, the staff concluded that granting of the requested exemption from the Type C testing requirements of 10 CFR Part 50, Appendix J, Option B, Section III.B, for LSS and NRS containment isolation valves that meet the licensees proposed criteria discussed and evaluated above, would pose no undue risk to public health and safety. As discussed in Section 20.2 of this SE, the staff found that the categorization process was not considered when the requirements of 10 CFR Part 50, Appendix J, Option B, Section III.B, were adopted and that it is in the public interest to grant an exemption from the special treatment requirements. This satisfies the special circumstance of 10 CFR 50.12(a)(2)(vi). Therefore, the staff determined that the exemptions should be granted from the Type C testing requirements of 10 CFR Part 50, Appendix J, Option B, Section III.B, as requested by STPNOC.

The following penetrations have been deleted from the LLRT program surveillance database. M-9, M-12, M-13, M-16, M-17, M-23, M-24, M- 25, M-26, M-27, M-28, M-29, M-30, M-34, M-36, M-38, M-39, M-40, M-45, M-56, M-57, M-58, M-61, M-68A, M-68C, M-68E, M-75, M-79, M-80A, M-80D, M-80E, M-80F, M-82A, M-82D, M-82E, M-85A, M- 85B, M-85E, M-86, M-88. The maintenance of exempt penetrations is described in Section 5.4.2.

In accordance with the exemption discussed above, Type C penetrations that are exempt from the Appendix J Option B program, values of zero (0) (i.e. max. path/min. path) may be entered for those exempted penetrations and components for the calculation of Type C leakage. At this time leakage rates for type C exempted penetrations will keep the last, As-Left, test values for conservatism instead of zero (0). Also the total number of type C penetrations used in the Calculation of Type "C" Leakage will remain the same for conservatism).

A review of the Type B and Type C test results from 2005 through 2014 for STP Unit 1 and 2006 through 2014 for STP Unit 2 has shown an exceptional amount of margin between the actual As-Found (AF) and As-left (AL) outage summations and the regulatory requirements as described below:

- The As-Found minimum pathway leak rate average for STP Unit 1 shows an average of 4.49% of 0.6 L_a with a high of 4.89% of 0.6 L_a or 0.029 L_a.
- The As-Left maximum pathway leak rate average for STP Unit 1 shows an average of 19.67% of 0.6 L_a with a high of 20.64% of 0.6 L_a or 0.124 L_a.
- The As-Found minimum pathway leak rate average for STP Unit 2 shows an average of 7.54% of 0.6 L_a with a high of 8.82% of 0.6 L_a or 0.053 L_a.
- The As-Left maximum pathway leak rate average for STP Unit 2 shows an average of 0.54% of 0.6 L_a with a high of 21.69% of 0.6 L_a or 0.130 L_a.

Tables 4.4.1-1 and 4.4.1-2 provide LLRT data trend summaries for STP since 2005 for Unit 1 and 2004 for Unit 2 and encompasses both previous ILRTs. This summary shows that there has been no As-Found failure that resulted in exceeding the Technical Specification 6.8.3.j limit of 0.6 L_a (455,050 sccm) and demonstrates a history of successful tests. The As-Found minimum pathway summations represent the high quality of maintenance of Type B and Type C tested components while the As-Left maximum pathway summations represent the effective management of the Containment Leakage Rate Testing Program by the program owner.

Table 4.4.1-1, Unit 1 Type B and C LLRT As-Found/As-Left Trend Summary

Refueling Outage	2005	2006	2008	2009	2011	2012	2014
AF Min Path (sccm)	18799.5	19123	20116	20773.1	21027.6	22237.5	20952.3
Fraction of L_a	2.48	2.52	2.65	2.74	2.77	2.93	2.76

Refueling Outage	2005	2006	2008	2009	2011	2012	2014
AL Max Path [1] (sccm)	83054.2	86542.2	89967.2	93440.3	93942.3	91053.2	88474.6
Fraction of L_a	10.95	11.41	11.86	12.32	12.39	12.01	11.67
AL Min Path (sccm)	18618.5	18942	19880.6	20588.6	21381.1	22549.0	21263.8
Fraction of La	2.45	2.50	2.62	2.71	2.82	2.97	2.80

Note 1: The AL Max Path summation contains additional margin provided by the continued inclusion of the last As-Left Maximum Pathway results for all exempted penetrations. For Unit 1 this value equals 50,787.8 sccm.

Table 4.4.1-2, Unit 2 Type B and C LLRT As-Found/As-Left Trend Summary

Refueling Outage	2004	2005	2007	2008	2010	2011	2013
AF Min Path (sccm)	33405.3	33144.5	31613.5	34613.5	40120	34261	33014
Fraction of L_a	4.40	4.37	4.17	4.56	5.29	4.52	4.35
AL Max Path [1] (sccm)	85234.5	93409.2	93606.2	98548.2	98717.2	88678.2	96007.2
Fraction of L_a	11.24	12.32	12.34	12.99	13.02	11.69	12.66
AL Min Path (sccm)	30597.3	30336.5	28805.5	31805.5	32911	31668	30421
Fraction of L_a	4.03	4.00	3.80	4.19	4.34	4.18	4.01

Note 1: The AL Max Path summation contains additional margin provided by the continued inclusion of the last As-Left Maximum Pathway results for all exempted penetrations. For Unit 2 this value equals 48,977.2 sccm.

The following, Tables 4.4.1-3 and 4.4.1-4, identify the components that have not demonstrated acceptable performance during the previous two outages for STP, Units 1 and 2 respectively:

Table 4.4.1-3, Unit 1 Type C LLRT Program Implementation Review

Component	As-found SCCM	Admin Limit SCCM	As-left SCCM	Cause of Failure	Corrective Action	Scheduled Interval
2012 1RE17						
None						
2014 1RE18						
M-43 Supplementary Containment Purge Valves MOV003	9352	7584	2288	Seat Leakage	Evaluated and repaired for continued service (1)	18 months. Extended interval not allowed

(1) While performing the as found LLRT on M-43 (Supplementary Containment Purge Supply) during 1RE18 the leakage rate exceeded the maintenance criteria. The maintenance criterion is 7584 sccm. The actual leak rate was 9352 sccm. CR 14-5133 was written to correct the condition. A Condition Report Engineering Evaluation (CREE 14-5133-2) was performed which concluded that an eighteen inch (18") blind flange meeting pipe specification PS004 can be used as an alternative temporary closure device until valve repairs are completed and acceptable Local Leak Rate Testing is performed prior to Mode 4. The temporary closure devise will prevent leakage of radioactive material in the event of a fuel handling accident inside containment.

While troubleshooting HCMOV0003 (M-43 ICIV) leakage, found significant deposits of red scale rust on the valve seat and disk. The penetration upstream (this valve is the ICIV for supply so the air from outside is pumped through it) looks like it had water condense inside it and the entire inside circumference is coated with scale rust. It is believed that the rust entered the valve and scored the soft Tefzel seat ring.

Maintenance was performed on 2V141THC0003 during 1RE18 (WAN 492938) and a post maintenance LLRT Surveillance was performed with satisfactory results prior to Mode 4.

Table 4.4.1-4, Unit 2 Type C LLRT Program Implementation Review

Component	As-found SCCM	Admin Limit SCCM	As-left SCCM	Cause of Failure	Corrective Action	Scheduled Interval
2011 2RE15						
M-44 Supplementary Containment Purge Exhaust HCFV9777	6625	6000	6625	Test duration (1)	Evaluated for continued service	18 month Extended interval not allowed
2013 Fall						
M-44 Supplementary Containment Purge Exhaust MOV0005	7285	6000	7285	Not Identified	Evaluated for continued service (2)	18 month Extended interval not allowed
2013 2RE16						
M-48 CVCS Charging CV0026	9442	4312	9442	Not Identified	Evaluated for continued service. (3)	30 month

(1) During performance of the LLRT on M-44 (Supplementary Containment Purge Exhaust) the leakage Rate was recorded at 7285 sccm. This leakage rate is below maintenance criteria (7584 sccm) but is over the administrative limit (6000 sccm) set by the LLRT program.

Typically the penetration is allowed two to three hours to stabilize. During this test the extended stabilization period was shortened due to scheduler considerations therefore have the higher than usual data which will show an increase in our trend. CR 11-23742 documents this above Admin limit test data and documents the extenuating conditions, short test window, associated with this test.

(2) During performance of the LLRT on M-44 (Supplementary Containment Purge Exhaust) the leakage Rate was recorded at 7285 sccm. This leakage rate is below maintenance criteria (7584 sccm) but is over the administrative limit (6000 sccm) set by the LLRT program.

Maintenance is scheduled on B2HCMOV0005 during 2RE16 (WAN 427451) and a post maintenance LLRT Surveillance will be performed.

Result of surveillance is SAT and meets TS surveillance requirements. Exceeding Admin Limits will initiate additional Eng. Evaluation of future preventive maintenance for better leakage results in the future if needed. Not an Operability issue.

(3) The leakage rate for check valve CV0026 was 9442 sccm. This leakage rate is greater than the administrative limit (4312 sccm) but less than the maintenance criteria limit (10,912 sccm) as identified in Addendum 4 of the program procedure 0PSP11-ZA-0005. Per the LLRT program requirements, this test is acceptable however a condition report is required per steps 9.2.10.2, 9.2.12.2, and 9.2.14 of the program procedure. LLRT result is within program controls and surveillance test result is acceptable.

CR Action 13-14450-1 written to identify corrective action to prevent recurrence of leak rate above Admin limit.

The percentage of the total number of Unit 1 and 2 Type B tested components (144) that are on 120-month extended performance-based test interval is 90%.

The percentage of the total number of Unit 1 and 2 Type C tested components (70) that are on 60 month extended performance-based test interval is 80%.

4.4.2 Maintenance of Exempted Penetrations

The purpose of the maintenance process for safety-related LSS and NRS SSCs is to establish the scope, frequency, and detail of maintenance activities necessary to support STPNOC's determination that these SSCs will remain capable of performing their safety-related functions under design-basis conditions. Preventive maintenance tasks are developed for active structures, systems, or components factoring in vendor recommendations. STPNOC may use an alternative to these recommendations if there is a technical basis that supports the functionality of the safety-related LSS and NRS SSCs. For an SSC in service beyond its designed life, STPNOC will have a technical basis to determine that the SSC will remain capable of performing its safety-related function(s).

The frequency and scope of predictive maintenance actions are established and documented considering vendor recommendations, environmental operating conditions, safety significance, and operating performance history. STPNOC may deviate from vendor recommendations where a technical basis supports the functionality of the safety-related LSS and NRS SSCs.

When an SSC deficiency is identified, it is documented and tracked through the Condition Reporting Program. The deficiency is evaluated to determine the corrective maintenance to be performed.

Following maintenance activities that affect the capability of a component to perform its safety-related function, post maintenance testing is performed to the extent necessary to provide reasonable confidence that the SSC is performing within expected parameters.

4.5 Nuclear Safety Advisory Letters (NSAL)

4.5.1 NSAL 11-05, Westinghouse LOCA Mass and Energy Release Calculation Issues," dated 07/25/2011 (Reference 25)

In NSAL 11-05, Westinghouse identified six issues that may adversely impact the LOCA mass and energy releases used in the containment pressurization analysis. STPNOC uses the Westinghouse provided mass and energy release to calculate containment conditions during a LOCA event. The NSAL identified five areas that may be significantly impacted by the LOCA issues. These areas are:

- Long Term Containment Peak Pressure Analysis
- Containment Peak Temperature Analysis
- Containment Equipment Qualification
- Containment Sump Temperature
- Ultimate Heat Sink

An operability review was performed and documented in CREE 11-12472-1. The results of the evaluation show that the condition does not result in any equipment being inoperable. However, the change in LOCA mass and energy results in a change to the peak containment pressure and temperature as presented in the UFSAR and TS 6.8.3.j. As a compensatory action (CR 11-12472-3), 0PSP11-ZA-0005, "Local Leakage Rate Test Calculations, Guidelines, and Program" has been revised to require a P_a of 43.2 psig to ensure sufficient margin until the UFSAR is updated to reflect the corrected value. Therefore, the determination of the condition is OPERABLE BUT NON-CONFORMING.

The operability review documented in CREE 11-12472-1 shows that there is not an operability concern. The analysis has been revised under CR 11-12472.

The event is due to errors in the vendor's analysis that are beyond the control of STPNOC. Therefore no actions to minimize the likelihood of occurrence are proposed.

4.5.2 NSAL 14-02, Westinghouse Loss-Of-Coolant Accident Mass And Energy Release Calculation Issue For Steam Generator Tube Material Properties, dated 07/25/2011 (Reference 35)

In NSAL 14-02, Westinghouse identified that the loss-of-coolant accident (LOCA) mass and energy (M&E) release analyses are sensitive to Energy stored in the reactor coolant system (RCS) metal, including the steam generator (SG) tubes. Recently, it was determined that the input modification program database and the input modification program preprocessor were using the density for stainless steel in determining the mass of the SG tubes and the specific heat (Cp) of stainless steel for the stored metal energy. Since all current Westinghouse-designed SGs use either alloy 600 or alloy 690 material for the SG tubes, there is a deviation from as-built plant parameters. Additionally, four plants for which Westinghouse has completed LOCA M&E calculations have non-Westinghouse-designed steam generators that have tubes manufactured from alloy 800 material.

To address this issue, Westinghouse has revised the LOCA mass and energy releases. The revised mass and energy releases were used in the analysis described in section 4.5.1, above, in addressing NSAL 11-05.

4.6 Information Notice (IN) 2010-12, "Containment Liner Corrosion"

This IN provides examples of containment liner degradation caused by corrosion. Concrete reactor containments are typically lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. The reactor containment is required to be operable as specified in plant technical specifications to limit the leakage of fission product radioactivity from the containment to the environment. The regulations at 10 CFR 50.55a, "Codes and Standards," require the use of Subsection IWE of ASME Section XI to perform inservice inspections of containment components. The required inservice inspections include periodic visual examinations and limited volumetric examinations using ultrasonic thickness measurements. The containment components include the steel containment liner and integral attachments for the concrete containment, containment personnel airlock and equipment hatch, penetration sleeves, moisture barriers, and pressure-retaining bolting. The NRC also requires licensees to perform leak rate testing of the containment pressure-retaining components and isolation valves according to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," as specified in plant technical specifications. This operating experience highlights the importance of good quality assurance, housekeeping and high quality construction practices during construction operations in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Corrosion to the containment liner is not a new industry issue. Programs and procedures are in-place to inspect the containment liner and would identify any areas subject to corrosion.

To date, STPNOC has not identified any degradation of the containment liner, penetrations, hatches, and their pressure retaining bolting.

4.7 Information Notice (IN) 2014-07, "Degradation of Leak Chase Channel Systems For Floor Welds Of Metal Containment Shell And Concrete Containment Metallic Liner"

The NRC issued this information notice (IN) to inform addressees of issues identified by the NRC staff concerning degradation of floor weld leak-chase channel systems of steel containment shell and concrete containment metallic liner that could affect leak-tightness and aging management of containment structures.

IN 2014-07 described the leak chase channel system as follows:

> Consists of steel channel sections that are fillet welded continuously over the entire bottom shell or liner seam welds and subdivided into zones, each zone with a test connection. Each test connection consists of a small carbon or stainless steel tube (less than 1-inch (2.5 centimeters)

diameter) that penetrates through the back of the channel and is seal-welded to the channel steel. The tube extends up through the concrete floor slab to a small steel access (junction) box embedded in the floor slab. The steel tube, which may be encased in a pipe, projects up through the bottom of the access box with a threaded coupling connection welded to the top of the tube, allowing for pressurization of the leak-chase channel.

IN 2014-07 describes a recessed box with a cover plate at floor level that allows for water to pool inside the recessed box and cause degradation.

There are 163 Containment Liner Weld Leak chase channels at STP, located on the minus 11 ft. elevation of the Reactor Containment Buildings. The majority of the Containment liner weld leak chase channel plates have been covered with an epoxy coating that is monitored during the assessment of the Service Level 1 Coatings inside the Reactor Containment Buildings. However, in support of the ILRT inspection the coatings can be removed from the floor plates to vent the containment liner weld leak chase channel as required per 0PSP11-IL-0007.

The general review of NRC Information Notice 2014-07 did not identify gaps in the IWE program. STP's Leak Chase Channels are considered inaccessible per USFAR section 3.8. In the development of the IWE program, the channels configuration met the definition of Exempt and Inaccessible from examination per IWE-1220 and IWE-1232 per ASME Section XI. The Containment's leak tightness has been established through successful completion of the ILRT testing.

To provide reasonable assurance that aging effects of the containment liner have been managed, engineering will perform a walk down of the 163 leak chase channel test connections, located on the minus 11 ft. elevation of the Reactor Containment Buildings during the 2RE17 and 1RE19 outages. This enhancement will insure that the components are still meeting the definition of inaccessible per IWE-1220 and IWE-1232 of ASME Section XI and that no degradation has taken place.

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

NRC Information Notice 92-20 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.

This is not applicable to STP in that installed bellows assemblies, which are also Containment isolation barriers, i.e., fuel transfer tube bellows, are of the single ply design. Reference Section 2.2.4.

4.9 Supplemental Inspections

In addition to the requirements of ASME Section XI, Subsections IWE and IWL, general visual inspections of the reactor containment are also required by TS 4.6.1.2, "Containment Leakage". This inspection is accomplished by the performance of a general visual inspection of the Reactor Containment using procedure 0PSP11-IL-0009, "Reactor Containment Building Visual Inspection."

The purpose of this procedure is to establish controls necessary to implement the inspection of the accessible interior and exterior surfaces of the Containment system, in accordance with 10CFR50.55a Appendix J, containment general visual inspection.

This procedure provides instructions for performing a visual inspection of the Reactor Containment Building (RCB) prior to any Type A test to verify that there is no obvious structural deterioration, which may affect either the containment structural integrity or leaktightness. This procedure must be conducted prior to each Type A test and during at least two other outages before the next Type A test if the Type A test interval is 10 years.

This procedure satisfies, in part, the requirements of Technical Specification 4.6.1.2.

This procedure satisfies the requirements of R.G. 1.163, "Performance-based Containment Leak-test Program", NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J", and ANSI/ANS-56.8, "Containment System Leakage Testing Requirements"

The inspections may be performed in conjunction or coordinated with the ASME Section XI, Subsection IWE/IWL required inspections.

4.10 Limitations and Conditions Applicable to NEI 94-01 Revision 2-A

The NRC staff found that the use of NEI TR 94-01, Revision 2, was acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT surveillance interval to 15 years, provided the following conditions as listed in Table 4.10-1 were satisfied.

Table 4.10-1, NEI 94-01 Revision 2-A Limitations and Conditions

Limitation/Condition (From Section 4.0 of SE)	STPNOC Response
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.)	STPNOC will utilize the definition in NEI 94-01 Revision 2-A, Section 5.0.
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.)	Reference Section 4.9 and Tables 4.1.3-1, 4.1.4-1 and 4.1.4-2 of this submittal.
The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3.)	Reference Sections 4.1.3 and 4.1.4 of this submittal.
The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4.)	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.
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 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.)	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, 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.

Limitation/Condition (From Section 4.0 of SE)	STPNOC Response
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 EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable. STP was not licensed under 10 CFR Part 52.

4.11 Evaluation of Risk Impact

4.11.1 Methodology

An evaluation has been performed to assess the risk associated with permanently extending the frequency of the Unit 1 and 2 Integrated Leak Rate Test (ILRT) from 10 years to 15 years. This surveillance frequency change will save one ILRT per unit after license extension, and will substantially reduce both station expense and critical path time during the associated outages. This risk assessment uses the guidance found in NEI 94-01, Revision 3-A (Reference 2), EPRI 1018243 (Reference 26), and Regulatory Guide (RG) 1.200 (Reference 5) as applied to ILRT interval extensions, and risk insights in support of a request for a change to the plant's licensing from RG 1.174 (Reference 4). NEI 94-01, Revision 3-A is used for guidance only and this assessment solely addresses ILRT extension and excludes Local Leakage Rate Testing (LLRT) extension. The Calvert Cliffs methodology (Reference 19) is used to estimate the likelihood and impact of undetected corrosion-induced leakage of the containment liner during the extended test interval.

This assessment calculated the effect on baseline population dose rate, the change in Large Early Release Frequency (LERF) and the change in Conditional Containment Failure Probability (CCFP). It also reviews the potential effects of containment liner corrosion on dose rate, LERF and CCFP, and the sensitivity of the results to the assumptions made in the liner corrosion analysis.

In the SER issued by NRC letter dated June 25, 2008 (Reference 9), the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE. Table 4.11.1-1 addresses each of the four limitations and conditions for the use of EPRI 1009325, Revision 2.

**Table 4.11.1-1 EPRI Report No.TR-1009325 Revision 2
Limitations and Conditions**

Limitation/Condition (From Section 4.2 of SE)	STPNOC Response
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension	The technical adequacy of the STP Unit 1 and Unit 2 PRA models are consistent with the requirements of Regulatory Guide 1.200 as is relevant to this ILRT interval extension.
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 3.2.4.5 of this SE. 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 CCFP should be defined as a value marginally greater than that accepted in a previous one-time ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.	EPRI Report No. 1009325, Revision 2-A, incorporates these population dose and CCFP acceptance guidelines, and these guidelines have been used for the STP plant specific assessments. The increase in population dose is 0.123 person-rem/year for Unit 1 and 2. The increase in CCFP is 0.87% for Units 1 and 2. Both Unit 1 and 2 prove to be below 1.5 percentage points and thus are considered to be small.
3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate accident case (accident case 3b) used by the licensees shall be 100 La instead of 35 La	EPRI Report No. 1009325, Revision 2-A, incorporated the use of 100 La 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 plant specific risk assessment.

Limitation/Condition (From Section 4.2 of SE)	STPNOC Response
4. A licensee amendment request (LAR) is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance.	For STP, containment over-pressure is NOT relied upon for emergency core cooling system (ECCS) performance.

4.11.2 Summary of Probabilistic Risk Assessment (PRA) Quality for Permanent 15-Year ILRT Extension

The STP PRA RV_72 (Reference 27) model, approved on 1/31/13, is a full-scope, at-power, integrated Level 1 and Level 2 PRA that applies to both STP Unit 1 and Unit 2. STP's PRA features a seismic PRA, fire PRA (including spatial interactions analysis), human reliability analysis, and detailed common cause modeling. STPNOC does not have separate internal, external or fire PRA models. Because the STP PRA is an at–power PRA, low power and shutdown events are not part of the scope. At-power scenarios bound low power and shutdown events in respect to challenges to containment, not only because the decay heat load is significantly reduced, but because the energy available to drive the accident scenario is much lower. The total effect of the change in ILRT periodicity, including internal, external and shutdown events, is thus bounded by all of the calculations in this assessment.

The PRA is maintained current, using a PRA configuration control program, in accordance with station procedures. It complies with station quality assurance procedures and requires software verification for the PRA quantification software (RISKMAN™), certification and qualification of software users, maintenance and update of the PRA, and performance of risk assessments. Periodic reviews and updates are made on a 3-year periodicity (including, at a minimum, updating equipment performance data, procedures, and modifications) by qualified personnel with independent reviews and approvals.

The STP PRA has a long history of independent technical reviews, industry peer reviews, and NRC technical reviews in support of many pilot efforts. STPNOC has used the PRA for risk-informed insights and applications since the mid-1980s. The NRC has previously reviewed the STP PRA in support of approving many risk-informed licensing applications, two recent examples of which are the Surveillance Frequency Control Program (Reference 28) and Risk-Managed Technical Specifications (RMTS) (Reference 30).

An industry peer review was performed on STP's PRA prior to the issuance of RG 1.200, Rev. 2. Since that time all findings and observations have been resolved and the PRA has been maintained in accordance with the PRA configuration control program, as discussed previously. The STP PRA model fully complies with Regulatory Guide (RG) 1.200, Revision 1.

The Fire PRA and Seismic PRA address all of the technical elements required by RG 1.200, Revision 1 and have been subjected to in-depth reviews to support license amendments for risk-informed applications, one example being RMTS. A detailed discussion of STP's PRA quality is contained in responses to NRC Requests for Additional Information (RAIs), Items 24 through 28, in an STPNOC letter to the NRC dated February 28, 2007 (Reference 29). Also contained in this letter is a detailed discussion of STP's Fire and Seismic PRAs prior to the issuance of RG 1.200, Rev.2. The Safety Evaluation Report approving the RMTS license amendment [24] indicated that the PRA was specifically reviewed for fire and external events and was found to be technically adequate.

STPNOC's PRA complies with Regulatory Guide (RG) 1.200, Rev. 2, with two exceptions. It does not comply with RG 1.200, Rev. 2 with respect to Fire PRA (e.g., new multiple spurious operation supporting requirements) and Seismic PRA requirements (e.g., incorporation of new seismic hazard curves). The Fire and Seismic PRAs that are integrated into the STP PRA model do not meet all of the requirements in the current ASME/ANS RA–S–2009 PRA Standard, as endorsed by RG 1.200, Rev. 2, at a Capability Category II level.

The risk assessment performed for this ILRT extension request is based on current Level 1 and Level 2 PRA model STP PRA RV_72. For this application, the accepted methodology involves a bounding approach to estimate the change in LERF, population dose and CCFP from extending the ILRT interval. Rather than modifying the PRA model itself, it involves the establishment of separate evaluations that use the plant's Core Damage Frequency (CDF), Level 2 Accident Progression Bins and Severe Accident Mitigation Alternatives (SAMA) analysis as inputs. The Level 2 Accident Progression Bins and the SAMA analysis are not expected to be significantly affected when the updated Fire and Seismic PRA requirements endorsed by RG 1.200, Rev. 2 are incorporated. Fire initiators have a low contribution to the probability of containment failure and STP is located in an area with low seismic activity. Therefore, the only plant-specific parameter that could significantly impact this assessment is CDF. The calculation of ΔCCFP is not sensitive to changing CDF. ΔLERF and the Change in Population Dose change in direct proportion to a change in CDF. It would take a significant change in CDF to challenge the conclusions reached by this assessment.

In conclusion, the STP PRA accurately reflects the as-built, as-operated facility and is technically adequate to evaluate and quantify the risk impact of changing the ILRT interval.

4.11.3 Summary of Plant-Specific Risk Assessment Results

The assessment of the plant risk associated with extending the Type A ILRT frequency from three in ten years to one in fifteen years concludes that:

- The increase in LERF, including the potential effect of liner corrosion is 5.27E-08/yr. Reg. Guide 1.174 (Reference 4) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10-6 /yr and increases in LERF below 10-7 /yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

- The change in total population dose risk, including the potential effect of liner corrosion, is 0.123 person-rem/yr. EPRI 1018243 (Reference 26) states that a very small population dose is defined as an increase of ≤1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The change in population dose is classified as "very small." This risk impact, when compared to other severe accident risks, is negligible.

- The increase in the conditional containment failure probability, including the potential effect of liner corrosion, is 0.87%. EPRI 1018243 (Reference 26) states that increases in CCFP of ≤1.5 percentage points are small and, therefore, this is classified as "a small increase."

The overall conclusion is that permanently increasing the ILRT interval to once every 15 years is acceptable since it represents a very small increase in the overall South Texas Project risk profile.

4.11.4 Previous Assessments

The NRC in NUREG-1493 has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from 3 per 10 years to 1 per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B or Type C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond 1 in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test integrity of the containment structure.

The findings for STP confirm these general findings on a plant-specific basis considering the severe accidents evaluated for STP, the STP containment failure modes, and the local population surrounding STP.

Details of the STP Unit 1 and 2, risk assessment are contained in Attachment 4 of this submittal.

5.0 No Significant Hazards Consideration

STP Nuclear Operating Company (STPNOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

The proposed amendment to the TS involves the extension of the STP, Units 1 and 2 Type A containment test interval to 15 years. The current Type A test interval of 120 months (10 years) would be extended on a permanent basis to no longer than 15 years from the last Type A test. Extensions of up to nine months (total maximum interval of 189 months for Type A tests) are permissible only for non-routine emergent conditions. The proposed extension does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. The change in dose risk for changing the Type A test frequency from once-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences for STP, of 0.123 person rem/yr for Unit 1 and Unit 2 using the EPRI guidance with the base case corrosion included. Therefore, this proposed extension does not involve a significant increase in the probability of an accident previously evaluated.

As documented in NUREG-1493, Type B and C tests have identified a very large percentage of containment leakage paths, and the percentage of containment leakage paths that are detected only by Type A testing is very small. The STP, Units 1 and 2 Type A test history supports this conclusion.

The integrity of the containment is subject to two types of failure mechanisms that can be categorized as: (1) activity based, and; (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the containment combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule, and TS requirements serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed extensions do not significantly increase the consequences of an accident previously evaluated.

The proposed amendment also deletes exceptions previously granted to allow one-time extensions of the ILRT test frequency for both Units 1 and 2. These exceptions were for activities that have already taken place so their deletion is solely an administrative action that has no effect on any component and no impact on how the units are operated.

Therefore, the proposed change does not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

The proposed amendment to the TS involves the extension of the STP, Unit 1 and 2 Type A containment test interval to 15 years. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

The proposed amendment also deletes exceptions previously granted to allow one-time extensions of the ILRT test frequency for both Units 1 and 2. These exceptions were for activities that would have already taken place by the time this amendment is approved; therefore, their deletion is solely an administrative action that does not result in any change in how the units are operated.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

The proposed amendment to TS 6.8.3.j involves the extension of the STP, Units 1 and 2 Type A containment test interval to 15 years. This amendment does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the TS Containment Leak Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained.

The proposed change involves only the extension of the interval between Type A containment leak rate tests for STP, Units 1 and 2. The proposed surveillance interval extension is bounded by the 15-year ILRT Interval currently authorized within NEI 94-01, Revision 2-A. Industry experience supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, TS and the Maintenance Rule serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by changes to the Type A and Type C test intervals.

The proposed amendment also deletes exceptions previously granted to allow one-time extensions of the ILRT test frequency for both Units 1 and 2. These exceptions were for activities that would have already taken place by the time this amendment is approved; therefore, their deletion is solely an administrative action and does not change how the units are operated and maintained. Thus, there is no reduction in any margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The proposed methodology satisfies the key principles of risk-informed decision making applied to changes to TSs as delineated in RG 1.177 and RG 1.174. The NRC staff, therefore, found that this guidance was 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 4.2 of the Safety Evaluation Report (SER).

Precedent

This request is similar in nature to the following license amendments to extend the Type A Test Frequency to 15 years, as previously authorized by the NRC:

- Nine Mile Point Nuclear Station, Unit 2 (Reference 21)
- Arkansas Nuclear One, Unit 2 (Reference 22)
- Palisades Nuclear Plant (Reference 23)
- Virgil C. Summer Nuclear Station, Unit 1 (Reference 24)

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 CONCLUSION

NEI 94-01, Revision 2-A, dated October 2008, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR Part 50, Appendix J, Option B. It incorporated the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 2-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. STPNOC is adopting the guidance of NEI 94-01, Revision 2-A, for the STP, Units 1 and 2, 10 CFR Part 50, Appendix J testing program plan.

Based on the previous ILRT tests conducted at STP, Units 1 and 2, it may be concluded that the permanent extension of the containment ILRT interval from 10 to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR Part 50, Appendix J and the overlapping inspection activities performed as part of the following STP, Units 1 and 2 inspection programs:

- Containment Inservice Inspection Program (IWE/IWL)
- Containment Coatings Assessment Program
- Technical Specification 4.6.1.2, Containment Leakage

This experience is supplemented by risk analysis studies, including the STP, Units 1 and 2, risk analysis provided in Attachment 4. The findings of the risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from 10 to 15 years results in a very small change to the STP, Units 1 and 2 risk profiles.

8.0 IMPLEMENTATION OF THE PROPOSED CHANGE

STPNOC requests approval of the proposed License Amendment by April 30, 2016, to be implemented within 90 days of the issuance of the license amendment.

9.0 REFERENCES

1. Regulatory Guide 1.163, Performance-Based Containment Leak-Test Program, September 1995.

2. NEI 94-01, Revision 3-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 2012.

3. NEI 94-01, Revision 2-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, October 2008.

4. Regulatory Guide 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.

5. Regulatory Guide 1.200, Revision 2, An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities, March 2009.

6. NEI 94-01, Revision 0, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 1995.

7. NUREG-1493, Performance-Based Containment Leak-Test Program, January 1995.

8. EPRI TR-104285, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, August 1994.

9. Letter from M. J. Maxin (NRC) to J. C. Butler (NEI), dated June 25, 2008, *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"* (TAC No. MC9663) ML081140105

10. Letter from S. Bahadur (NRC) to B. Bradley (NEI), dated June 8, 2012, 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) ML110890314

11. Letter from T. W. Alexion (NRC) to W. T. Cottle dated May 19, 1995. SOUTH TEXAS PROJECT, UNITS 1 AND 2 -AMENDMENT NOS. 75 AND 64 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. M91830 AND M91831) ML021300263

12. Letter from T. W. Alexion (NRC) to W. T. Cottle dated September 7, 1995. SOUTH TEXAS PROJECT, UNITS 1 AND 2 -AMENDMENT NOS. 80 AND 69 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. M92517 AND M92518) ML021330525

13. Letter from T. W. Alexion (NRC) to W. T. Cottle dated August 13, 1996. SOUTH TEXAS PROJECT, UNITS 1 AND 2 - AMENDMENT NOS. 84 AND 71 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. M94536 AND M94538) ML021300572

14. Letter from J. A. Zwolinski (NRC) to W. T. Cottle dated August 3, 2001. SOUTH TEXAS PROJECT, UNITS 1 AND 2 - SAFETY EVALUATION ON EXEMPTION REQUESTS FROM SPECIAL TREATMENT REQUIREMENTS OF 10 CFR PARTS 21, 50, AND 100 (TAC NOS. MA6057 AND MA6058) ML011990368

15. Letter from M. C. Thadani (NRC) to W. T. Cottle dated April 12, 2002. SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS APPROVING UPRATED CORE THERMAL POWER AND REVISING THE ASSOCIATED TECHNICAL SPECIFICATIONS (TAC NOS. MB2899 AND MB2903) ML020800263

16. Letter from M. C. Thadani (NRC) to W. T. Cottle dated July 18, 2002. SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS ON

EQUIPMENT HATCH OPEN DURING REFUEL OPERATIONS (TAC NOS. MB3587 AND MB3591) ML021430328

17. EPRI Report 1003102, "Guideline on Nuclear Safety-Related Coatings", Revision 1 (formerly TR-109937).

18. Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

19. Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002 ML020920100

20. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325, EPRI, Palo Alto, CA: 2008.

21. Letter from R. V. Guzman (NRC) to S. L. Belcher (NMP), dated March 30, 2010, Nine Mile Point Nuclear Station, Unit No. 2 - Issuance of Amendment RE: Extension of Primary Containment Integrated Leakage Rate Testing Interval (TAC No. ME1650) ML100730032

22. Letter from N. K. Kalyanam (NRC) to Vice President, Operations (ANO), dated April 7, 2011, Arkansas Nuclear One, Unit No. 2 - Issuance of Amendment RE: Technical Specification Change to Extend Type A Test Frequency to 15 Years (TAC No. ME4090) ML110800034

23. Letter from M. L. Chawala (NRC) to Vice President, Operations (PNP), dated April 23, 2012, Palisades Nuclear Plant - Issuance of Amendment to Extend the Containment Type A Leak Rate Test Frequency to 15 Years (TAC No. ME5997) ML120740081

24. Letter from S. Williams (NRC) to T. D. Gatlin (VCSNS), dated February 5, 2014, Virgil C. Summer Nuclear Station, Unit 1 - Issuance of Amendment Extending Integrated Leak Rate Test Interval (TAC No. MF1385) ML13326A204

25. NSAL 11-05, "Westinghouse LOCA Mass and Energy Release Calculation Issues," dated 07/25/2011.

26. EPRI 1018243, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, Revision 2-A of 1009325

27. STP_RV72 PRA, STI 33651684, South Texas Project Level I and II Approved PRA, 1/31/13.

28. Letter from M. C. Thadani (NRC) to E. D. Halpin dated October 31, 2008. SOUTH TEXAS PROJECT, UNITS 1 AND 2 -ISSUANCE OF AMENDMENTS (188 & 175) TO RELOCATE SURVEILLANCE TEST INTERVALS TO LICENSEE-CONTROLLED PROGRAM (RISK-INFORMED INITIATIVE 5-b) (TAC NOS. MD7058 AND MD7059) ML082830172

29. NOC-AE-07002112, STP Units 1 & 2 Response to NRC Requests for Additional Information on STPNOC Proposed Risk Managed Technical Specifications (TAC Nos. MD 2341 & MD 2342) dated February 28, 2007 ML070670369

30. Letter from M. C. Thadani (NRC) to J. J. Sheppard dated July 13, 2007. SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (179 & 166) RE: BROAD-SCOPE RISK-INFORMED TECHNICAL SPECIFICATIONS AMENDMENTS (TAC NOS. MD2341 AND MD2342) (AE-NOC-07001652) ML071780186

31. Letter from M. C. Thadani (NRC) to W. T. Cottle dated September 17, 2002. SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISING THE APPENDIX J INTEGRATED LEAK RATE TESTING INTERVAL (TAC NOS. MB2897 AND MB2901) ML022410163

32. Letter from M. Thadani (NRC) to W. T. Cottle dated January 7, 2003. SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EXTENSION OF THE INTERVALS BETWEEN OPERABILITY TESTS OF THE NORMAL AND SUPPLEMENTARY CONTAINMENT PURGE VALVES (TAC NOS. MB4048 and MB4049) ML030130435

33. Letter from M. C. Thadani (NRC) to E.D. Halpin dated January 30, 2009. SOUTH TEXAS PROJECT, UNITS 1 AND 2 -ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATION 3.6.1.3, "CONTAINMENT AIR LOCKS" (TAC NOS. MD8156 AND MD8157) ML083640080

34. Letter from J. A. Zwolinski (NRC) to W. T. Cottle dated August 3, 2001. SOUTH TEXAS PROJECT, UNITS 1 AND 2 - SAFETY EVALUATION ON EXEMPTION REQUESTS FROM SPECIAL TREATMENT REQUIREMENTS OF 10 CFR PARTS 21, 50, AND 100 (TAC NOS. MA6057 AND MA6058) ML011990368

35. NASL 14-02, Westinghouse Loss-Of-Coolant Accident Mass And Energy Release Calculation Issue For Steam Generator Tube Material Properties, dated 07/25/2011

Attachment 2

Markup of Technical Specification Page 6.8.3.j

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 ≤ 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 Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program", dated September 1995. The current ten-year interval between performance of the integrated leakage 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).

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

(continued)

Attachment 3

Typed Technical Specification Page 6.8.3.j

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(0) 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)

Attachment 4

PRA Evaluation Permanent ILRT Extension Risk Assessment

PRA-14-015 Rev. 0
STP Risk Assessment for Extending ILRT Interval to 15 Years

STI: 34037063	**0PGP05-ZE-0001**	**Rev. 3**	
PRA Analyses/Assessments			
Form 1	Analysis/Assessment Package Cover Sheet		

PRA ANALYSIS/ASSESSMENT

Analysis/Assessment Sequence Number: PRA-14-015 **Rev.:** 0

Ref. CR#: 15-848 (if applicable)

Subject:

STP Risk Assessment for Extending ILRT Interval to 15 Years

Description:

The purpose of this analysis is to perform a quantitative and qualitative risk evaluation to support a License Amendment Request (LAR) to change the Integrated Leak Rate Test (ILRT) surveillance frequency from 10 years to 15 years. This assessment is developed using EPRI guidance.

Documents Used by this Analysis/Assessment:

EPRI Report 1018243, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325

NEI 94-01, Revision 3-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J

Documents Supported by this Analysis/Assessment:

Preparer: Brian Ratté / B. R. Ratte Date: 1/15/15

(ESP Cert. 9287)

Reviewer: Coral Betancourt / C. Betancourt Date: 1/15/15

(ESP Cert. 9287)

Additional Reviews Performed by:

Shawn Rodgers / Shawn Rodgers Date: 1-15-2015

/ Date:

Approved: R. F. Dunn / RF Dunn Date: 1/15/15

Manager, Nuclear Fuel & Analysis or Supervisor, Risk Management

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STP Risk Assessment for Extending ILRT Interval to 15 Years

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PRA-14-015 Rev. 0
STP Risk Assessment for Extending ILRT Interval to 15 Years

PURPOSE AND SCOPE

The purpose of this analysis is to perform an evaluation to assess the risk associated with permanently extending the frequency of the Unit 1 and 2 Integrated Leak Rate Test (ILRT) from 10 years to 15 years. This surveillance frequency change will save one ILRT per unit after license extension, and will substantially reduce both station expense and critical path time during the associated outages. This risk assessment uses the guidance found in NEI 94-01, Revision 3-A [1], EPRI 1018243 [2], Regulatory Guide (RG) 1.200 [3] as applied to ILRT interval extensions, and risk insights in support of a request for a change to the plant's licensing from RG 1.174 [4]. NEI 94-01, Revision 3-A is used for guidance only and this assessment solely addresses ILRT extension and excludes Local Leakage Rate Testing (LLRT) extension. The Calvert Cliffs methodology [5] is used to estimate the likelihood and impact of undetected corrosion-induced leakage of the containment liner during the extended test interval.

This assessment will calculate the effect on baseline population dose rate, the change in Large Early Release Frequency (LERF) and the change in Conditional Containment Failure Probability (CCFP). It also reviews the potential effects of containment liner corrosion on dose rate, LERF and CCFP, and the sensitivity of the results to the assumptions made in the liner corrosion analysis.

BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage rate was less than limiting containment leakage rate of 1 La (La is allowable leakage). La (percent/24 hours) is the maximum allowable leakage rate at test pressure (Pa). At STP, Pa is 41.2 psig and La is 0.3% per day [22].

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program" [6], provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals" [7]. Refer to Addendum 1 for additional perspective on these references and other risk assessment documents associated with containment leakage.

To complement EPRI report TR-104285, which only considered changes to the ILRT testing intervals based on population dose, EPRI report 1018243 (Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325) was developed that considers population dose, Large Early Release Frequency (LERF) and Containment Conditional Failure Probability (CCFP). EPRI report 1018243 indicates that, in general, the risk impact associated with ILRT interval extensions for intervals up to fifteen years is small. However, a plant specific confirmatory analysis is required. The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from containment leak rate testing. In that analysis, it was determined, for a representative PWR plant (i.e., Surry), that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. It is necessary to show that extending the ILRT interval will have a similarly small increase in risk from containment isolation failures for STP.

PRA-14-015 Rev. 0
STP Risk Assessment for Extending ILRT Interval to 15 Years

The guidance provided in Appendix H of EPRI 1018243 for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. EPRI 1018243 outlines the method used to evaluate the risk impact of the proposed ILRT interval changes.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment.

Visual examinations must be performed in the outage during which the ILRT is conducted, and during at least three other outages between ILRTs. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency. Note that STP has an exemption [8] which allows some mechanical penetrations to be excluded from periodic LLRTs. To be exempted, containment isolation valves must meet specific criteria that is listed in Updated Final Safety Analysis Report (UFSAR) Table 13.7-1. Exemption is only allowed for components that have both low safety significance and a negligible impact on containment leak-tight integrity.

Note that the ten-year interval between performance of the integrated leakage rate (Type A) test, beginning September 24, 1991, for Unit 2 and March 10, 1995, for Unit 1, was extended to 15 years (a one-time change). The license amendment was requested in August of 2001 [9,10] and was supported by a risk assessment in March of 2002 [11] using the NEI template that was in effect at that time. The current interval is ten years.

METHOD

NEI 94-01, Revision 3A and EPRI 1018243 will be used to evaluate the risk impact of changing the ILRT frequency. Only the ILRT frequency is evaluated for risk impact and other inspections and tests will continue at their current intervals. A simplified bounding analysis approach is used to evaluate changing the ILRT frequency to 15 years. The analysis uses results from the Level 2 analysis of core damage scenarios from the current STP_RV72 PRA model. The SAMA analysis for the Units 1 and 2 License Extension license amendment request [12] is used to obtain population dose. The method is summarized below and further described in the analysis section.

- The first three steps of the methodology calculate the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The population dose at the new interval is calculated by multiplying the base population dose (from the SAMA analysis) by the change in the probability of a containment leakage event for the affected Core Damage Frequency (CDF) end states. The metrics associated with absolute population dose change and change in population dose as a percentage of the total dose are both calculated. The range of incremental population dose increases for previously submitted one-time ILRT interval extensions is from ≤0.01 to 0.2 person-rem/yr or 0.002 to 0.46% of the total accident dose (note that the one-time submittals used a large leak magnitude of 35La and this methodology uses 100La, which will result in larger calculated doses). The total doses for the spectrum of all accidents (NUREG-1493, Figure 7-2) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a very small population dose is defined as an increase of ≤ 1.0 person-rem/yr or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals.

- The fourth step in the methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF (containment overpressure is not credited for the STP Emergency Core Cooling System (ECCS) Net Positive Suction Head (NPSH)), the change in LERF suffices as the quantitative basis for a risk-informed decision per current NRC practice, namely Regulatory Guide 1.174. Regulatory Guide 1.174 defines "very small changes" in risk as an increase in LERF of less than 10^{-7} per reactor year. Regulatory Guide 1.174 also defines "small changes" in risk as a change in LERF of less than 10^{-6} per reactor year. Very small changes do not require a calculation of the total LERF, but small changes will require total LERF to be less than 10^{-5} per reactor year to be considered by the NRC.
- The fifth step calculates the change in containment failure probability. The NRC has previously accepted similar calculations, referred to as Conditional Containment Failure Probability (CCFP), as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. As such, this step suffices as the remaining basis for a risk-informed decision per Regulatory Guide 1.174. Changes of up to 1.1% have been accepted by the NRC for one-time extensions of ILRT intervals. An increase in CCFP of ≤1.5% is assumed to be small.
- The sixth and final step assesses the impact of extended intervals on containment liner corrosion in a sensitivity analysis. As before, the metrics associated with absolute population dose change and change in population dose as a percentage of the total dose are both calculated. The change in LERF and CCFP are also calculated and all are compared to the previously defined criteria.

Consistent with other industry containment leak risk assessments, the South Texas Project assessment uses LERF and ΔLERF in accordance with the risk acceptance guidance of RG 1.174. Changes in population dose and CCFP are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.

ASSUMPTIONS

- STP RV_72 model assumptions are valid and remain unchanged for this assessment
- The representative containment leakage for EPRI Accident Class 1 (Intact Containment) sequences is 1La. Class 1 represents sequences where containment remains intact.
- Class 3a and 3b account for increased leakage due to Type A containment inspection failures.
- The representative containment leakage for EPRI Accident Class 3a sequences is 10La, based on the methodology approved for Indian Point Unit 3. Class 3a represents Small Early Release.
- The representative containment leakage for EPRI Accident Class 3b sequences is 100La, based on the guidance in EPRI Report 1018243. Class 3b is conservatively categorized as Large Early Release, based on previously approved methodology.
- Containment bypass scenarios are not affected by changes in ILRT frequency.
- The reliability of containment isolation valves is not affected by changes in ILRT frequency.
- The assumptions associated with the corrosion sensitivity analysis are listed in Step 6.

ANALYSIS

PRA Description

The STP PRA RV_72 [13] model, approved on 1/31/13, is a full-scope, at-power, integrated Level 1 and Level 2 PRA that applies to both STP Unit 1 and Unit 2. STP's PRA features a seismic PRA, fire PRA (including spatial interactions analysis), human reliability analysis, and detailed common cause modeling. STP does not have separate internal, external or fire PRA models. Because the STP PRA is an at–power PRA, low power and shutdown events are not part of the scope. At-power scenarios bound low power and shutdown events in respect to challenges to containment, not only because the decay heat load is significantly reduced, but because the energy available to drive the accident scenario is much lower. The total effect of the change in ILRT periodicity, including internal, external and shutdown events, is thus bounded by all of the calculations in this assessment.

The PRA is maintained current, using a PRA configuration control program, in accordance with station procedures. It complies with station quality assurance procedures and requires software verification for the PRA quantification software (RISKMAN™), certification and qualification of software users, maintenance and update of the PRA, and performance of risk assessments. Periodic reviews and updates are made on a 3-year periodicity (including, at a minimum, updating equipment performance data, procedures, and modifications) by qualified personnel with independent reviews and approvals.

The STP PRA has a long history of independent technical reviews, industry peer reviews, and NRC technical reviews in support of many pilot efforts. STPNOC has used the PRA for risk-informed insights and applications since the mid-1980s. The NRC has previously reviewed the STP PRA in support of approving many risk-informed licensing applications, two recent examples of which are the Surveillance Frequency Control Program [20] and Risk-Managed Technical Specifications (RMTS) [21].

An industry peer review was performed on STP's PRA prior to the issuance of RG 1.200, Rev. 2. Since that time all findings and observations have been resolved and the PRA has been maintained in accordance with the PRA configuration control program, as discussed previously. The STP PRA model fully complies with Regulatory Guide (RG) 1.200, Revision 1.

The Fire PRA and Seismic PRA address all of the technical elements required by RG 1.200, Revision 1 and have been subjected to in-depth reviews to support license amendments for risk-informed applications, one example being RMTS. A detailed discussion of STP's PRA quality is contained in responses to NRC Requests for Additional Information (RAIs), Items 24 through 28, in an STP letter to the NRC dated February 28, 2007 [23]. Also contained in this letter is a detailed discussion of STP's Fire and Seismic PRAs prior to the issuance of RG 1.200, Rev.2. The Safety Evaluation Report approving the RMTS license amendment [24] indicated that the PRA was specifically reviewed for fire and external events and was found to be technically adequate.

STP's PRA complies with Regulatory Guide (RG) 1.200, Rev. 2, with two exceptions. It does not comply with RG 1.200, Rev. 2 with respect to Fire PRA (e.g., new multiple spurious operation supporting requirements) and Seismic PRA requirements (e.g., incorporation of new seismic hazard curves). The Fire and Seismic PRAs that are integrated into the STP PRA model do not meet all of the requirements in the current ASME/ANS RA–S–2009 PRA Standard, as endorsed by RG 1.200, Rev. 2, at a Capability Category II level.

The risk assessment performed for this ILRT extension request is based on current Level 1 and Level 2 PRA model STP PRA RV_72. For this application, the accepted methodology involves a bounding approach to estimate the change in LERF, population dose and CCFP from extending the ILRT interval. Rather than modifying the PRA model itself, it involves the establishment of separate evaluations that use the plant's CDF, Level 2 Accident Progression Bins and SAMA analysis as inputs. The Level 2 Accident Progression Bins and the SAMA analysis are not expected to be significantly affected when the updated Fire and Seismic PRA requirements endorsed by RG 1.200, Rev. 2 are incorporated. Fire initiators have a low contribution to the probability of containment failure and STP is located in an area with low seismic activity. Therefore, the only plant-specific parameter that could significantly impact this assessment is CDF. The calculation of ΔCCFP is relatively insensitive to changes in CDF. ΔLERF and the Change in Population Dose change in direct proportion to a change in CDF. It would take a significant change in CDF to challenge the conclusions reached by this assessment.

In conclusion, the STP PRA accurately reflects the as-built, as-operated facility and is technically adequate to evaluate and quantify the risk impact of changing the ILRT interval.

STEP 1: Quantify the baseline (three-year ILRT frequency) risk in terms of frequency per reactor year for the EPRI accident classes of interest.

The EPRI accident classes, as described in EPRI 1018243, are listed in the following table:

Table 1-1: EPRI Accident Classes

Accident Class	Description	Frequency
1	This sequence class consists of all core damage accident progression bins for which the containment remains intact with negligible leakage. Class 1 sequences arise from those core damage sequences where containment isolation is successful and long-term containment heat removal capability is available.	CDF_{Intact} - $F_{Class\ 3a}$ - $F_{Class\ 3b}$ Where: CDF_{Intact} = the core damage frequency for intact containment sequences from the plant-specific PRA.
2	This group consists of all core damage accident progression bins for which a pre-existing leakage due to failure to isolate the containment occurs. These sequences are dominated by failure to close of large (>2 inches [5.1 cm] in diameter) containment isolation valves.	$F_{Class\ 2} = PROB_{large\ CI} * CDF_{Total}$ Where: $PROB_{large\ CI}$ = random containment large isolation failure probability (large valves) CDF_{Total} = total plant-specific core damage frequency, which is obtained from plant specific PRA.
3a	All core damage accident progression bins with a pre-existing leakage in the containment structure in excess of normal leakage up to 10 La (small leakage). $PROB_{Class\ 3a}$ is a function of ILRT test interval. La = Allowable Leakage	$PROB_{Class\ 3a}$ * CDF $PROB_{Class\ 3a}$ = the probability of small pre-existing containment leakage in excess of design allowable but less than 10 La. CDF = total plant-specific core damage frequency, which is obtained from plant specific PRA.

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Accident Class	Description	Frequency
3b	All core damage accident progression bins with a pre-existing leakage in the containment structure in excess of normal leakage of 100 La (large leakage). $PROB_{Class\ 3b}$ is a function of ILRT test interval.	$PROB_{Class\ 3b}$ * CDF $PROB_{Class\ 3b}$ = the probability of large (100 La) pre-existing containment leakage. CDF = total plant-specific core damage frequency, which is obtained from plant specific PRA.
4	This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type B test components occurs. Because these failures are detected by Type B tests and their frequency is very low compared with the other classes, this group is not evaluated any further. The frequency for Class 4 sequences is subsumed into Class 7, where it contributes insignificantly.	N/A
5	This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type C test components occurs. Because these failures are detected by Type C tests and their frequency is very low compared with the other classes, this group is not evaluated any further. The frequency for Class 5 sequences is subsumed into Class 7, where it contributes insignificantly.	N/A
6	This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage, due to failure to isolate the containment, occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution, typically resulting in a failure to close smaller containment isolation valves. All other failure modes are bounded by the Class 2 assumptions. This accident class is not evaluated further.	N/A
7	This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (for example, H_2 combustion and direct containment heating): $F_{Class\ 7}$ can be determined by subtracting the intact, bypass (Class 8) and loss of isolation CDFs from the total CDF. These end states include containment failure.	$F_{Class\ 7} = CDF_{CFL} + CDF_{CFE}$ Where: CDF_{CFE} = the core damage frequency resulting from accident sequences that lead to early containment failure. CDF_{CFL} = the core damage frequency resulting from accident sequences that lead to late containment failure.

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Accident Class	Description	Frequency
8	This group consists of all core damage accident progression bins in which containment bypass occurs. Each plant's PRA is used to determine the containment bypass contribution. Contributors to bypass events include ISLOCA events and SGTRs with an unisolated steam generator. The magnitude of bypass releases is plant-specific and is typically considerably larger (two or more orders of magnitude) than releases expected for leakage events. The containment structure will not impact the release magnitude for this event class.	$CDF_{ISLOCA} + CDF_{Unisolated\ SGTR}$

The EPRI accident classes do not directly correlate to the STP Level 2 release bins and so each release bin must be reviewed and assigned to an EPRI accident class. Note that the total of the Level 2 release bins frequency is not equal to the total CDF of 6.0624E-06 and is low by approximately 4.6%. This is due to rounding and truncation in the bin calculations. A scaling factor of 1.048 was added to adjust the total bin frequency up to the total CDF and the result is shown in the rightmost column of Table 1-2. The scaling factor is determined by dividing the overall CDF by the sum of the release bin frequencies.

Table 1-2: STP_RV72 PRA Model Level 2 Release Bins

Bin	Containment Failure Mode	Containment Spray Recirc	RCS Press at Vessel Breach (psia)	Frequency (per Yr)	Adjusted Frequency (per Yr)
BYPASS	Bypass, Small	No	Any	2.67E-08	2.79E-08
CICV	Pre-existing, small	No	No breach	4.21E-08	4.41E-08
INTACT1	None	N/A	Any	1.68E-07	1.76E-07
INTACT2	None	N/A	No breach	2.59E-06	2.72E-06
ISGTR	Bypass, Large	N/A	Medium	3.67E-07	3.84E-07
R01	Early, Large	No	≥2000	2.72E-10	2.85E-10
R01U	Early, Large	No	≥2000	3.00E-09	3.15E-09
R02	Early, Large	Available	≥2000	0.00E+00	0.00E+00
R02U	Early, Large	Available	≥2000	0.00E+00	0.00E+00
R03	Early, Large	No	≥200	1.62E-10	1.70E-10
R03U	Early, Large	No	≥200	1.59E-09	1.66E-09
R04	Early, Large	Available	≥200	0.00E+00	0.00E+00
R04U	Early, Large	Available	≥200	0.00E+00	0.00E+00
R05	Early, Small	No	≥200	0.00E+00	0.00E+00
R05L	Early, Small	No	≥200	0.00E+00	0.00E+00
R05LU	Early, Small	No	≥200	0.00E+00	0.00E+00
R05S	Pre-existing, small	No	≥200	4.03E-09	4.22E-09
R05SL	Pre-existing, small	No	≥200	1.16E-10	1.22E-10
R05SLU	Pre-existing, small	No	≥200	8.00E-08	8.39E-08
R05SU	Pre-existing, small	No	≥200	8.14E-07	8.53E-07
R05U	Early, Small	No	≥200	2.76E-11	2.90E-11
R06	Early, Small	Available	≥200	0.00E+00	0.00E+00
R06L	Early, Small	Available	≥200	0.00E+00	0.00E+00
R06LU	Early, Small	Available	≥200	0.00E+00	0.00E+00
R06S	Pre-existing, small	Available	≥200	2.11E-08	2.21E-08
R06SL	Pre-existing, small	Available	≥200	5.00E-10	5.24E-10
R06SLU	Pre-existing, small	Available	≥200	3.84E-09	4.03E-09
R06SU	Pre-existing, small	Available	≥200	3.38E-08	3.54E-08
R06U	Early, Small	Available	≥200	0.00E+00	0.00E+00
R07	Early, Small	No	<200	0.00E+00	0.00E+00
R07L	Early, Small	No	<200	0.00E+00	0.00E+00
R07LU	Early, Small	No	<200	0.00E+00	0.00E+00

Bin	Containment Failure Mode	Containment Spray Recirc	RCS Press at Vessel Breach (psia)	Frequency (per Yr)	Adjusted Frequency (per Yr)
R07S	Pre-existing, small	No	<200	2.60E-11	2.73E-11
R07SL	Pre-existing, small	No	<200	0.00E+00	0.00E+00
R07SLU	Pre-existing, small	No	<200	1.34E-08	1.41E-08
R07SU	Pre-existing, small	No	<200	1.40E-07	1.46E-07
R07U	Early, Small	No	<200	3.28E-12	3.44E-12
R08	Early, Small	Available	<200	0.00E+00	0.00E+00
R08L	Early, Small	Available	<200	0.00E+00	0.00E+00
R08LU	Early, Small	Available	<200	0.00E+00	0.00E+00
R08S	Pre-existing, small	Available	<200	7.56E-09	7.92E-09
R08SL	Pre-existing, small	Available	<200	2.10E-10	2.20E-10
R08SLU	Pre-existing, small	Available	<200	1.95E-10	2.04E-10
R08SU	Pre-existing, small	Available	<200	9.86E-09	1.03E-08
R08U	Early, Small	Available	<200	0.00E+00	0.00E+00
R09	Late, Large	No	≥200	8.90E-09	9.33E-09
R09U	Late, Large	No	≥200	8.22E-08	8.61E-08
R10	Late, Large	Available	≥200	2.87E-10	3.01E-10
R10U	Late, Large	Available	≥200	0.00E+00	0.00E+00
R11	Late, Large	No	<200	6.00E-09	6.28E-09
R11U	Late, Large	No	<200	2.11E-07	2.21E-07
R12	Late, Large	Available	<200	0.00E+00	0.00E+00
R12U	Late, Large	Available	<200	0.00E+00	0.00E+00
R13	Late, Small	No	≥200	2.79E-08	2.92E-08
R13U	Late, Small	No	≥200	3.68E-07	3.86E-07
R14	Late, Small	Available	≥200	2.48E-08	2.60E-08
R14U	Late, Small	Available	≥200	6.42E-09	6.73E-09
R15	Late, Small	No	<200	6.48E-11	6.79E-11
R15U	Late, Small	No	<200	7.18E-07	7.53E-07
R16	Late, Small	Available	<200	1.01E-10	1.05E-10
R16U	Late, Small	Available	<200	4.58E-10	4.79E-10
VSEQ	Bypass, Large	N/A	Any	5.21E-12	5.46E-12
Total:				5.79E-06	6.06E-06

The Table 1-2 frequencies, as reported by PRA RV_72, may be found in the Computer Input/Output section.

Note that Table 1-2 columns 1 through 4 are based on Table 4-2 of the STP_REV7 Level 2 Containment Event Tree (CET) notebook [15].

- "U" at the end of a bin name indicates that the core debris is not cooled and so is "uncovered."
- "L" indicates that there is a Large Late Release in addition to a Small Early Release
- "S" at the end of a bin name indicates a pre-existing containment failure.
    - Containment failure is categorized as "pre-existing" if it occurs prior to core damage (i.e., failure to isolate containment) and as "early" if it occurs within 4 hours after vessel breach.

The probability of a large pre-existing leak in the reactor containment building is not modeled in the South Texas Project Probabilistic Risk Assessment. The results for LERF are dominated by sequences caused by a phenomenon called Induced Steam Generator Tube Rupture (ISGTR) which occurs when the secondary side of the steam generators dries out after a core damage event with the reactor coolant system intact at high pressure. High temperature coolant circulates through the Reactor Coolant System, heating up the steam generator tubes to the point of failure. The Induced Steam Generator Tube Rupture sequences are primarily caused by core damage scenarios that involve loss of all station AC power (Station Blackout). The Integrated Leakage Rate Test does not test this pathway through the steam generators. The dominant cause of containment bypass is failure of the supplementary containment purge to isolate during an accident sequence. This sequence is also not affected by Integrated Leakage Rate Testing. The Small Early Release Frequency (SERF) group includes the potential for a small preexisting leak. A small containment failure existing prior to core damage is the most important contributor to SERF. Most of this contribution is from steam generator tube ruptures, but a significant portion derives from preexisting leaks in the containment building.

Containment overpressure is not required to provide Emergency Core Cooling System (ECCS) Net Positive Suction Head (NPSH) in the South Texas Project containment design and CDF is thus unaffected by this parameter.

Review of the STP PRA Level 2 model results in the grouping in Table 1-3 for the various EPRI accident classes. Note that some of the EPRI accident classes are divided into LERF and non-LERF for the purpose of calculating the total frequency of those bins that represent LERF.

Table 1-3: STP Release Categories and Corresponding EPRI Accident Class

STP Major Release Group	STP Release Categories	Corresponding EPRI Release Bin Grouping (in bold)	EPRI Acc Class	EPRI Acc Sub-Class	Notes
REL I (LERF)	R01, R01U, R02, R02U, R03, R03U, R04, R04U	*Large, early* **containment failures due to accident phenomenon,** at any RCS pressure	7	7 LERF	All are shown as Large Early Release in Table 4-2 of the Level 2 CET Notebook
REL II (SERF)	R05, R05L, R05U, R05LU, R06, R06L, R06U, R06LU	*Small, early* **containment failures due to accident phenomenon,** with RCS pressure >200 psia	7	7 non-LERF	All are shown as Small Early Release at High RCS Pressure in Table 4-2 of the CET Notebook. Some are also shown as Large Late Release (indicated by an "L" in the bin name) and are included here for conservatism.
REL II (SERF)	R07, R07L, R07U, R07LU, R08, R08L, R08U, R08LU	*Small, early* **containment failures due to accident phenomenon,** with RCS pressure <200 psia	7	7 non-LERF	All are shown as Small Early Release at Low RCS Pressure in Table 4-2 of the Level 2 CET Notebook. Some are also shown as Large Late Release and are included here for conservatism.
REL II (SERF)	CICV, R05S, R05SL, R05SLU, R05SU, R06S, R06SL, R06SLU, R06SU	*Small, early* **containment failures due to failure to isolate,** with RCS pressure >200 psia	2	2 non-LERF	CICV and R05SU have a total of a 3-inch diameter leak path (see MAAP analysis in STP_REV7 Level 2 Accident Progression notebook [16]). These are conservatively placed in EPRI class 2 (see Table 1-1 definition) because they are leaks greater than 2-inches in diameter that are due to failures of containment isolation (the "S" indicates leakage that occurs prior to core damage).

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STP Major Release Group	STP Release Categories	Corresponding EPRI Release Bin Grouping (in bold)	EPRI Acc Class	EPRI Acc Sub-Class	Notes
REL II (SERF)	R07S, R07SL, R07SLU, R07SU, R08S, R08SL, R08SLU, R08SU	*Small, early* **containment failures due to failure to isolate**, with RCS pressure <200 psia	2	2 non-LERF	R07SU has a total of a 3-inch diameter leak path (see MAAP analysis in STP_REV7 Level 2 Accident Progression notebook). These are conservatively placed in EPRI class 2 (see Table 1-1 definition) because they are leaks greater than 2-inches in diameter that are due to failures of containment isolation (the "S" indicates leakage that occurs prior to core damage).
REL III (LATE)	R09, R09U, R10, R10U, R11, R11U, R12, R12U	*Large, late* **containment failures due to accident phenomenon**	7	7 non-LERF	All are shown are described as Large Late Release at Low RCS Pressure in Table 4-2 of the Level 2 CET Notebook. Containment failure occurs after core damage.
REL III (LATE)	R13, R13U, R14, R14U, R15, R15U, R16, R16U	*Small, late* **containment failures due to accident phenomenon**	7	7 non-LERF	All are shown are described as Small Late Release at Low RCS Pressure in Table 4-2 of the Level 2 CET Notebook. Containment failure occurs after core damage.
REL I (Cntnmt Bypass)	VSEQ, ISGTR	*Large* **containment bypass** (includes interfacing-systems LOCAs, induced SGTRs and unscrubbed faulted S/Gs)	8	8 LERF	These large containment bypasses are described in the Level 2 CET notebook.
REL III (Cntnmt Bypass)	BYPASS	Small **containment bypass** (includes scrubbed faulted S/Gs)	8	8 non-LERF	This bin accounts for small containment bypass, as described in the Level 2 CET notebook.
REL IV (Cntnmnt Intact)	INTACT1, INTACT2	Long term containment integrity (**Intact Containment**)	1	1	These two bins are described in the Level 2 CET notebook as specifically being created for sequences where containment remains intact.

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The EPRI Accident Sub-Class frequencies are calculated in the following table by reorganizing Table 1-3 and summing the bin frequencies from Table 1-2:

Table 1-4: EPRI Accident Sub-Classes and Total Frequencies

EPRI Accident Sub-Class	STP PRA Level 2 Bins	EPRI Accident Sub-Class Frequency/yr
1	INTACT1, INTACT2	2.89E-06
2 non-LERF	CICV, R05S, R05SL, R05SLU, R05SU, R06S, R06SL, R06SLU, R06SU, R07S,R07SL, R07SLU, R07SU, R08S, R08SL, R08SLU, R08SU	1.23E-06
7 non-LERF	R05, R05L, R05U, R05LU, R06, R06L, R06U, R06LU, R07, R07L, R07U, R07LU, R08, R08L, R08U, R08LU, R09, R09U,R10, R10U, R11, R11U, R12, R12U, R13, R13U, R14, R14U, R15, R15U, R16, R16U	1.52E-06
7 LERF	R01, R01U, R02, R02U, R03, R03U, R04, R04U	5.26E-09
8 non-LERF	BYPASS	2.79E-08
8 LERF	VSEQ, ISGTR	3.84E-07

All that remains is to calculate the EPRI Class 3a (small leakage of 10 La) and 3b (large leakage of 100 La) frequency (see Table 1-1 for details), as follows:

Industry information indicates that there have been two ILRT failures out of 217 tests. These failures were small leaks and are those that could only have been identified with an ILRT. It is appropriate, and conservative, to utilize the maximum likelihood estimate (arithmetic average) (2/217 = 0.0092) for the class 3a (10 La) distribution.

Class 3a (10 La) represents small pre-existing containment leakage that would only be detected by an ILRT.

Class 3a probability is assigned the maximum likelihood estimate of 2 failures in 217 tests, which is 0.0092. Whenever used in this assessment, CDF refers to the STP_RV72 baseline CDF of 6.0624E-06. For brevity only two decimal places are shown, but all calculations use frequencies calculated to four decimal places.

Class 3a frequency = CDF * Class 3a leakage probability

Class 3a frequency = 6.06E-06/yr * 0.0092 = **5.59E-08/yr**

No large leaks have been identified with an ILRT. Using the definition of a large early release as being greater than 35 La (from Reference [17]), there are no containment leakage events that could result in a large early release in the current dataset. The zero failures are based on the combined ILRT database (NUMARC and NEI surveys [18, 19] and other sources), in which the results of 217 ILRTs have been documented. (It is conservatively estimated that over 400 ILRTs have been performed in the U.S. nuclear industry. The 217 ILRTs that have been documented are used in this submittal and analysis.)

Class 3b (100 La) represents large pre-existing containment leakage that would only be detected by an ILRT and is conservatively assumed to result in a large early release.

With zero failed events, a variety of statistical methods is available to estimate a failure rate. However, for the purposes of this analysis, it is believed that the Jeffreys Non-Informed Prior provides a reasonable balance between conservatism in light of uncertainty while still meeting the intent of RG 1.174.

Class 3b (LERF) probability uses the Jeffries non-informed prior: $\dfrac{\text{Number of Failures} + 1/2}{\text{Number of Tests} + 1}$

(Number of Failures + 1/2) / (Number of Tests + 1) = (0 + 0.5)/(217+1) = 0.0023

The methodology employed for determining LERF (Class 3b frequency) could conservatively multiply the CDF by the failure probability for this class of accident. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with the postulated large Type A containment leakage path. These contributors can be removed from Class 3b by multiplying the Class 3b probability by only that portion of CDF that may be impacted by Type A leakage.

Class 3b frequency = (CDF – (frequency of sequences directly causing LERF)) * Class 3b leakage probability

Note: Only EPRI Accident Class 7 and 8 LERF frequencies are subtracted from the baseline CDF. Sequences that could never cause a LERF are not subtracted out and this is conservative.

Class 3b frequency = (CDF – (Class 7 LERF + Class 8 LERF)) * 0.0023

Class 3b frequency = (6.06E-06/yr – (5.26E-09/yr + 3.84E-07/yr)) * 0.0023 = **1.30E-08/yr**

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The class 3a and 3b frequencies are now subtracted from the Class 1 frequency so that total frequency will remain the same.

Class 1 Frequency (revised) = Class 1 Frequency – (Class 3a Frequency + Class 3b Frequency)

Class 1 Frequency (revised) = 2.89E-06/yr – (5.59E-08/yr + 1.30E-08/yr) = **2.83E-06/yr**

The revised EPRI Accident Class Table is Below with LERF and non-LERF bins combined:

Table 1-5: EPRI Accident Classes and Total Frequencies

EPRI Accident Class	STP PRA Level 2 Bins	EPRI Accident Class Frequency/yr
1	INTACT1, INTACT2	2.83E-06
2	CICV, R05S, R05SL, R05SLU, R05SU, R06S, R06SL, R06SLU, R06SU, R07S,R07SL, R07SLU, R07SU, R08S, R08SL, R08SLU, R08SU	1.23E-06
3a	None	5.59E-08
3b	None	1.30E-08
7	R01, R01U, R02, R02U, R03, R03U, R04, R04U, R05, R05L, R05U, R05LU, R06, R06L, R06U, R06LU, R07, R07L, R07U, R07LU, R08, R08L, R08U, R08LU, R09, R09U,R10, R10U, R11, R11U, R12, R12U, R13, R13U, R14, R14U, R15, R15U, R16, R16U	1.53E-06
8	BYPASS, VSEQ, ISGTR	4.12E-07
Total		6.06E-06

STEP 2: Develop the baseline population dose (person-rem, from the plant PRA or IPE, or calculated based on leakage) for the applicable accident classes.
The current baseline model of record, STP_RV72 [13] will be used for all calculations. Where the SAMA analysis and associated PRA Level 3 analysis refer back to previous model STP_RV6 [14], the current model's frequencies will be substituted.

The Severe Accident Management Analysis (SAMA) report for STP License Extension [3] was used to obtain the 2050 population dose for representative accident classes. The population dose figures are by MACCS2 analysis and are based on a projected population of 455,418 within 50 miles of the plant in the year 2050.

Table 2-1: SAMA Report Population Dose Bin Assignments

STP SAMA Release Category and Representative Bin	Population Dose at 50 Miles	Bins Assigned to this Category by this Assessment (Using Table 1-3)
Group I (ISGTR)	1.36E+06	ISGTR, VSEQ, R01, R01U, R02, R02U, R03, R03U, R04, R04U
Group II (R05SU)	5.12E+05	R05S, R05SL, R05SLU, R05SU, R06S, R06SL, R06SLU, R06SU, R05, R05L, R05U, R05LU, R06, R06L, R06U, R06LU
Group II (CICV)	2.12E+05	CICV
Group II (R07SU)	7.50E+05	R07S, R07SL, R07SLU, R07SU, R08S, R08SL, R08SLU, R08SU, R07, R07L, R07U, R07LU, R08, R08L, R08U, R08LU
Group III (R15U)	1.49E+05	R15, R15U, R16, R16U
Group III (R13U)	2.85E+05	R13, R13U, R14, R14U
Group III (R11U)	4.25E+05	R09, R09U, R10, R10U, R11, R11U, R12, R12U
Group III (Bypass)	2.22E+06	BYPASS
Group IV (Intact)	1.70E+04	INTACT1, INTACT2

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The population dose for the SAMA release category and representative bin was assigned to each of the bins shown in Table 2-1 and the dose risk for each is calculated as shown in Table 2-2:

Table 2-2: STP Population Dose and Dose Risk by Bin

STP BIN	Adjusted Frequency (per Yr) (Table 1-2)	Population Dose (Person-Rem) (Table 2-1)	Dose Risk = Dose * Freq (Person-Rem/Yr)
BYPASS	2.79E-08	2.22E+06	6.20E-02
CICV	4.41E-08	2.12E+05	9.35E-03
INTACT1	1.71E-07 (Note 1)	1.70E+04	2.91E-03
INTACT2	2.65E-06 (Note 1)	1.70E+04	4.51E-02
ISGTR	3.84E-07	1.36E+06	5.22E-01
R01	2.85E-10	1.36E+06	3.88E-04
R01U	3.15E-09	1.36E+06	4.28E-03
R02	0.00E+00	1.36E+06	0.00E+00
R02U	0.00E+00	1.36E+06	0.00E+00
R03	1.70E-10	1.36E+06	2.31E-04
R03U	1.66E-09	1.36E+06	2.26E-03
R04	0.00E+00	1.36E+06	0.00E+00
R04U	0.00E+00	1.36E+06	0.00E+00
R05	0.00E+00	5.12E+05	0.00E+00
R05L	0.00E+00	5.12E+05	0.00E+00
R05LU	0.00E+00	5.12E+05	0.00E+00
R05S	4.22E-09	5.12E+05	2.16E-03
R05SL	1.22E-10	5.12E+05	6.24E-05
R05SLU	8.39E-08	5.12E+05	4.29E-02
R05SU	8.53E-07	5.12E+05	4.37E-01
R05U	2.90E-11	5.12E+05	1.48E-05
R06	0.00E+00	5.12E+05	0.00E+00
R06L	0.00E+00	5.12E+05	0.00E+00
R06LU	0.00E+00	5.12E+05	0.00E+00
R06S	2.21E-08	5.12E+05	1.13E-02
R06SL	5.24E-10	5.12E+05	2.68E-04
R06SLU	4.03E-09	5.12E+05	2.06E-03
R06SU	3.54E-08	5.12E+05	1.81E-02
R06U	0.00E+00	5.12E+05	0.00E+00
R07	0.00E+00	7.50E+05	0.00E+00
R07L	0.00E+00	7.50E+05	0.00E+00

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STP BIN	Adjusted Frequency (per Yr) (Table 1-2)	Population Dose (Person-Rem) (Table 2-1)	Dose Risk = Dose * Freq (Person-Rem/Yr)
R07LU	0.00E+00	7.50E+05	0.00E+00
R07S	2.73E-11	7.50E+05	2.05E-05
R07SL	0.00E+00	7.50E+05	0.00E+00
R07SLU	1.41E-08	7.50E+05	1.06E-02
R07SU	1.46E-07	7.50E+05	1.10E-01
R07U	3.44E-12	7.50E+05	2.58E-06
R08	0.00E+00	7.50E+05	0.00E+00
R08L	0.00E+00	7.50E+05	0.00E+00
R08LU	0.00E+00	7.50E+05	0.00E+00
R08S	7.92E-09	7.50E+05	5.94E-03
R08SL	2.20E-10	7.50E+05	1.65E-04
R08SLU	2.04E-10	7.50E+05	1.53E-04
R08SU	1.03E-08	7.50E+05	7.75E-03
R08U	0.00E+00	7.50E+05	0.00E+00
R09	9.33E-09	4.25E+05	3.96E-03
R09U	8.61E-08	4.25E+05	3.66E-02
R10	3.01E-10	4.25E+05	1.28E-04
R10U	0.00E+00	4.25E+05	0.00E+00
R11	6.28E-09	4.25E+05	2.67E-03
R11U	2.21E-07	4.25E+05	9.39E-02
R12	0.00E+00	4.25E+05	0.00E+00
R12U	0.00E+00	4.25E+05	0.00E+00
R13	2.92E-08	2.85E+05	8.33E-03
R13U	3.86E-07	2.85E+05	1.10E-01
R14	2.60E-08	2.85E+05	7.40E-03
R14U	6.73E-09	2.85E+05	1.92E-03
R15	6.79E-11	1.49E+05	1.01E-05
R15U	7.53E-07	1.49E+05	1.12E-01
R16	1.05E-10	1.49E+05	1.57E-05
R16U	4.79E-10	1.49E+05	7.14E-05
VSEQ	5.46E-12	1.36E+06	7.42E-06

Note 1: These frequencies are adjusted from those in Table 1-2 to account for the subtraction of the Class 3a and 3b frequencies from Class 1.

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Obtain the Average Population Dose for each EPRI Class as Follows:

1. Sum up the Level 2 bins assigned to each EPRI Accident Class in Table 2-2 to obtain the dose risk.
2. As each EPRI Accident Class contains more than one STP bin, a frequency-weighted dose is used to represent EPRI Class 1, 2, 7 and 8 (see below).
3. Class 3a population dose is assumed to be 10 times more than Class 1.
4. Class 3b population dose is assumed to be 100 times more than Class 1.
5. The dose risk for Class 3a and 3b are obtained by dividing the Average Population Dose by the EPRI Accident Class Frequency.

Frequency-weighted dose is simply the sum of dose risk divided by the sum of the frequencies of all of the STP bins in the corresponding EPRI Class.

An example of how the frequency-weighted average population dose is calculated is as follows:

From Table 2-3, Class 8 is comprised of Release Category Group Level 2 Bins "Bypass", "VSEQ" and "ISGTR." From Table 1-5, the frequency for Class 8 is 4.12E-07/yr, which is the sum of the STP PRA Level 2 bin frequencies. From Table 2-2, the Dose risk is calculated for each bin and Bypass is 6.20E-02, VSEQ is 7.42E-06 and ISGTR is 5.22E-01.

The frequency-weighted average population dose for Class 8 is:

= (Bypass + VSEQ + ISGTR dose risk) / (Bypass + VSEQ + ISGTR frequency)

= (6.20 E-02 + 7.42E-06 + 5.22E-01) person-rem/yr / (2.79E-08 + 5.46 E-12 + 3.84E-07) per yr

= (5.84 E-01 person-rem/yr) / (4.12E-07 /yr) = **1.42E+06 person-rem**

Table 2-3: Population Dose and Dose Risk by EPRI Accident Class

EPRI Accident Class	STP PRA Level 2 Bins (Table 1-5)	Class Frequency (per yr)	Population Dose Risk (person-rem/yr)	Weighted Average Population Dose at 50 miles (person-rem)
1	INTACT1, INTACT2	2.83E-06	4.80E-02	1.70E+04
2	CICV, R05S, R05SL, R05SLU, R05SU, R06S, R06SL, R06SLU, R06SU, R07S,R07SL, R07SLU, R07SU, R08S, R08SL, R08SLU, R08SU	1.23E-06	6.57E-01	5.36E+05
3a	N/A	5.59E-08	9.50E-03	1.70E+05 (Note 1)
3b	N/A	1.30E-08	2.21E-02	1.70E+06 (Note 2)
7	R01, R01U, R02, R02U, R03, R03U, R04, R04U, R05, R05L, R05U, R05LU, R06, R06L, R06U, R06LU, R07, R07L, R07U, R07LU, R08, R08L, R08U, R08LU, R09, R09U,R10, R10U, R11, R11U, R12, R12U, R13, R13U, R14, R14U, R15, R15U, R16, R16U	1.53E-06	3.84E-01	2.51E+05
8	BYPASS, VSEQ, ISGTR	4.12E-07	5.84E-01	1.42E+06
Total	N/A	6.06E-06	1.71	N/A

Note 1: Class 3a (10 La) population dose is calculated by multiplying the intact containment (EPRI Class 1) population dose times 10.

Note 2: Class 3b (100 La) population dose is calculated by multiplying the intact containment (EPRI Class 1) population dose times 100.

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STEP 3: Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
The Population Dose and Base Case Frequency and Dose Risk is taken directly from Table 2-3. Changing the test interval potentially affects Class 3a and 3b frequency as they are pre-existing leakage that would be detected by an ILRT. The frequency is conservatively assumed to increase linearly with time.

- An increase from 3 inspections every 10 years to 1 inspection every 10 years is an increase of 10/3 = 3.33 in the Class 3a and 3b frequency.
- An increase from 3 inspections every 10 years to 1 inspection every 15 years is an increase of 15/3 = 5 in the Class 3a and 3b frequency.
- The increase in Class 3a and 3b frequency is subtracted from the Class 1 frequency to maintain total frequency equal to the plant's CDF.
- Dose Risk is recalculated for EPRI Class 1, 3a and 3b
- The increase in Total Dose Risk reflects the change in population dose.

Table 3-1: Effect of Extension on Dose Risk

		Base Case (3 per 10 yrs)		Interval Extended to 10 yrs		Interval Extended to 15 yrs	
EPRI Class	Avg Pop. Dose at 50 miles (Person-Rem)	Freq (Per year)	Dose Risk (Person-Rem/Yr)	Freq (Per year)	Dose Risk (Person-Rem/Yr)	Freq (Per year)	Dose Risk (Person-Rem/Yr)
1	1.70E+04	2.83E-06	4.80E-02	2.66E-06	4.53E-02	2.55E-06	4.33E-02
2	5.36E+05	1.23E-06	6.57E-01	1.23E-06	6.57E-01	1.23E-06	6.57E-01
3a	1.70E+05	5.59E-08	9.50E-03	1.86E-07	3.17E-02	2.79E-07	4.75E-02
3b	1.70E+06	1.30E-08	2.21E-02	4.34E-08	7.37E-02	6.51E-08	1.11E-01
7	2.51E+05	1.53E-06	3.84E-01	1.53E-06	3.84E-01	1.53E-06	3.84E-01
8	1.42E+06	4.12E-07	5.84E-01	4.12E-07	5.84E-01	4.12E-07	5.84E-01
Total	N/A	6.06E-06	1.71E+00	6.06E-06	1.78E+00	6.06E-06	1.83E+00
Change in Dose Risk			N/A		7.10E-02		1.22E-01
Percent Change in Dose Risk			N/A		4.16%		7.14%

From EPRI 1018243, a very small population dose is defined as an increase of ≤ 1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals.

The change in dose risk is less than 1.0 person-rem per year for both the 10-year and 15-year interval, and so it meets the definition of "a very small population dose" as defined in EPRI 1018243.

STEP 4: Determine the risk impact in terms of the change in LERF.
The risk associated with extending the ILRT interval involves a potential that a core damage event that normally would result in only a small radioactive release from containment will result in a large release due to an undetected leak path existing during the extended interval. As discussed in References [6] and [7], only Class 3 sequences have the potential to result in early releases if a pre-existing leak were present. Late releases are excluded regardless of the size of the leak because late releases are not, by definition, LERF events. The frequency of class 3b sequences is used as a measure of LERF, and the change in LERF (ΔLERF) is determined by the change in class 3b frequency. Refer to Regulatory Guide 1.174 [4] for LERF acceptance guidelines. ΔLERF is determined using the equation below, where the "frequency of class 3b frequency x" is the frequency of the EPRI accident class 3b for the ILRT interval of interest and the "frequency of class 3b baseline" is defined as the EPRI accident class 3b frequency for ILRTs performed on a three-per-10-years basis. All of these frequencies are contained in Table 3-1. Note that all frequencies are calculated to four decimal places and rounded to two decimal places.

ΔLERF = (frequency of class 3b new interval x) - (frequency of class 3b baseline)

ΔLERF 10 year interval = (4.34E-08/yr) - (1.30E-08/yr) = 3.04E-08/yr

ΔLERF 15 year interval = (6.51E-08/yr) - (1.30E-08/yr) = 5.20E-08/yr

The 10-year and 15-year ΔLERF fall into the category of "a very small change in risk" per the Regulatory Guide 1.174 definition of an increase in LERF of less than 1.0E-07 per reactor year.

STEP 5: Determine the risk impact in terms of the change in Conditional Containment Failure Probability.
The conditional containment failure probability (CCFP) is defined as the probability of containment failure given the occurrence of a core damage accident, which can be expressed as:

CCFP = [1 - (frequency that results in no containment failure) / CDF] * 100%

CCFP = [1 - (frequency class 1 + frequency class 3a) / CDF] * 100%

CCFP Change (increase) = (CCFP at interval x) – (CCFP at baseline interval), expressed as a percentage point change.

As above, all of these frequencies may be found in Table 3-1.

CCFP baseline = [1 - (2.83E-06/yr + 5.59E-08/yr) / 6.06E-06/yr] * 100% = 52.47%

CCFP 10 years = [1 - (2.66E-06/yr + 1.86E-07/yr) / 6.06E-06/yr] * 100% = 52.98%

CCFP 15 years = [1 - (2.55E-06/yr + 2.79E-07/yr) / 6.06E-06/yr] * 100% = 53.33%

ΔCCFP 10 year interval = 52.98% – 52.47% = 0.50%

ΔCCFP 15 year interval = 53.33% – 52.47% = 0.86%

From EPRI 1018243, a small increase in CCFP is defined as an increase of ≤ 1.5%.

The change in CCFP for both the 10-year and 15-year interval falls into the definition of "a small increase" as defined in EPRI 1018243.

STEP 6: Account for the potential effects of liner corrosion and evaluate sensitivity to the liner corrosion analysis assumptions.

This analysis presents an estimate of the likelihood and risk implications of corrosion induced leakage of steel containment liners being undetected during extended ILRT test intervals. The methodology employed is taken from the Calvert Cliffs liner corrosion analysis [5]. It is important to note that the corrosion analysis is a sensitivity case that represents the first 15-year extension. It is possible that for some slow corrosion mechanisms, such as embedment of debris during initial containment construction, the probability of leakage will increase over a long period of time. However, these mechanisms are generally very slow and have a limited potential for the development of large leakage pathways before detection. The Calvert Cliffs analysis is performed for a concrete cylinder and dome with a concrete basemat, each with a steel liner. South Texas Project has a similar containment type. The following approach is used to determine the change in likelihood, due to extending the ILRT interval, of detecting corrosion of the steel liner and thus the potential change in risk. Consistent with the Calvert Cliffs analysis, the following are addressed:

- Differences between the containment basemat and the containment cylinder and dome
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

The assumptions used in this sensitivity study are consistent with the Calvert Cliffs methodology and include the following:

- A half failure is assumed for the basemat concealed liner corrosion due to lack of identified failures.
- Two corrosion events are used to estimate the liner flaw probability. These events, one at North Anna Unit 2 and the other at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.
- The estimated historical flaw probability is limited to 5.5 years to reflect the years since September 1996 when 10CFR50.55a started requiring visual inspections. Additional success data were not used to limit the aging impact of the corrosion issue, even though inspections were being performed prior to this data (and have been performed since the timeframe of the Calvert Cliffs analysis) and there has been no evidence that additional corrosion issues were identified.
- The likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat in the Calvert Cliffs analysis. These values were determined from an assessment of the probability versus containment pressure, corresponding to the ILRT pressure. For South Texas Project, the containment is tested to 41.2 (+3, -0) psig, and the cylinder and dome failure probability at this pressure is much less than this value, as indicated by curves in the plant's IPE, Section 4.7. Conservative failure probabilities of 1% and 0.1% are used for the cylinder and dome and basemat, respectively.
- The likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less than that in the containment cylinder and dome region.
- A 5% visual inspection detection failure likelihood, given that the flaw is visible, and a total detection failure likelihood of 10% are used. To date, all liner corrosion events have been detected through visual inspection.
- All non-detectable failures are assumed to result in large early releases. This approach is conservative and avoids detailed analysis of containment failure timing and operator recovery actions. That is, the probability of all non-detectable failures from the corrosion sensitivity analysis are added to the EPRI Class 3b (and subtracted from EPRI Class 1).
- See the Table 6-1 notes for additional assumptions.

Table 6-1: Probability of Undetected Containment Leakage

Step	Description	Containment Cylinder and Dome	Containment Basemat
1	Historical Steel Liner Flaw Likelihood	Events: 2	Events: 0 (assume half a failure)
	Failure Data:(1)	2 / (70 * 5.5) = 5.2E-3	0.5 / (70 * 5.5) = 1.3E-3
2	Age-Adjusted Steel Liner Flaw Likelihood (2)	Year / Failure Rate 1 2.1E-03 Avg 5-10 5.2E-03 15 1.4E-02 15-year avg = 6.44E-3	Year / Failure Rate 1 5.1E-04 Avg 5-10 1.3E-03 15 3.6E-03 15-year avg = 1.61E-3
3	Flaw Likelihood at 3, 10, and 15 years (3a)	0.71% (1 to 3 years) 4.14% (1 to 10 years) 9.66% (1 to 15 years) (3b)	0.18% (1 to 3 years) 1.03% (1 to 10 years) 2.41% (1 to 15 years) (3c)
4	Likelihood of Breach in Containment Given Steel Liner Flaw (4)	1%	0.1%
5	Visual Inspection Detection Failure Likelihood	10% (5a)	100% (5b)
6	Likelihood of Undetected Containment Leakage (Steps 3*4*5)	0.00071% (at 3 years) 0.71% *1% * 10% 0.00414% (at 10 years) 4.14% * 1% * 10% 0.00966% (at 15 years) 9.66% * 1% * 10%	0.00018% (at 3 years) 0.18% * 0.1% * 100% 0.00103% (at 10 years) 1.03% * 0.1% * 100% 0.00241% (at 15 years) 2.41% * 0.1% * 100%

Notes:

(1) Containment location specific (consistent with Calvert Cliffs analysis).

(2) During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the fifth to tenth year is set to the historical failure rate (consistent with Calvert Cliffs analysis).

(3) (a) Uses age-adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs).

(3) (b) Note that the Calvert Cliffs analysis presents the delta between three and 15 years of 8.7% to utilize in the estimation of the ΔLERF value. For this analysis, however, the values are calculated based on three-, 10-, and 15-year intervals, consistent with the desired presentation of the results.

(3) (c) Note that the Calvert Cliffs analysis presents the delta between three and 15 years of 2.2% to utilize in the estimation of the ΔLERF value. For this analysis, however, the values are calculated based on the three-, 10-, and 15-year intervals, consistent with the desired presentation of the results.

(4) The failure probability of the cylinder and dome is assumed to be 1%, and basemat is 0.1% as compared to 1.1% and 0.11% in the Calvert Cliffs analysis.

(5) (a) 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through cylinder but could be detected by ILRT). All flaws have been detected through visual inspection. Five percent visible failure detection is a conservative assumption.

(5) (b) Cannot be visually inspected.

The total likelihood of corrosion-induced, undetected containment failure, which is the sum of Table 6-1, Step 6 for the containment cylinder, dome and basemat is summarized below:

Table 6-2: Likelihood of Undetected Containment Leakage Due to Corrosion Effects

Time	Probability of Undetected Leakage		
	Containment Cylinder and Dome	Containment Basemat	Total
3 years	0.00071%	0.00018%	0.00089% (8.9E-06)
10 years	0.00414%	0.00103%	0.00517% (5.2E-05)
15 years	0.00966%	0.00241%	0.01207% (1.2E-04)

The corrosion sensitivity is calculated from the undetected containment leakage probability as follows:

1. As discussed in Step 1 of this analysis, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with the postulated large Type A containment leakage path in EPRI Accident Class 3b. Subtracting these from the total CDF will yield the CDF associated with accidents that do not independently cause a large early release:

   CDF – (Class 7 LERF + Class 8 LERF) = 6.06E-06/yr – (5.26E-09/yr + 3.84E-07/yr) = 5.67E-06/yr

2. The increase in the base case (3 year) Class 3b frequency due to corrosion-induced concealed flaws is: 5.67E-06/yr * 8.9E-06 = 5.05E-11/yr
3. From Table 1-5, the base case Class 3b frequency = 1.30E-08/yr
4. The base case Class 3b frequency is added to the corrosion-induced concealed flaw frequency to provide the new Class 3b frequency: 1.30E-08/yr + 5.05E-11/yr = 1.31E-08/yr
5. The change in Class 3b frequency is subtracted from the Class 1 frequency to obtain the new Class 1 frequency, and thus maintaining the same total frequency (which is equal to CDF):

   2.83E-06/yr – 5.05E-11/yr = 2.83E-06/yr

6. The average population dose for each Class is multiplied by the frequency to provide the dose risk for each class and summed to provide the total dose risk. See Table 6-3 for the results.
7. The change in dose risk, with corrosion effects included, is calculated to compare to the base case.
8. The 10-year and 15-year corrosion sensitivity is calculated in a similar manner. All results are summarized in Table 6-3.

Table 6-3: Steel Liner Corrosion Sensitivity Summary

Summary of STP Total Risk for Undetected Corrosion													
		Base Case (3 Per 10 Years)				Interval Extended to 10 Years				Interval Extended to 15 Years			
		Without Corrosion		With Corrosion		Without Corrosion		With Corrosion		Without Corrosion		With Corrosion	
Class	Avg Pop. Dose	Freq	Dose Risk	Freq	Dose Risk	Freq	Dose Risk	Freq	Dose Risk	Freq	Dose Risk	Freq	Dose Risk
1	1.70E+04	2.83E-06	4.80E-02	2.83E-06	4.80E-02	2.66E-06	4.53E-02	2.66E-06	4.53E-02	2.55E-06	4.33E-02	2.54E-06	4.33E-02
2	5.36E+05	1.23E-06	6.57E-01	1.23E-06	6.57E-01	1.23E-06	6.57E-01	1.23E-06	6.57E-01	1.23E-06	6.57E-01	1.23E-06	6.57E-01
3a	1.70E+05	5.59E-08	9.50E-03	5.59E-08	9.50E-03	1.86E-07	3.17E-02	1.86E-07	3.17E-02	2.79E-07	4.75E-02	2.79E-07	4.75E-02
3b	1.70E+06	1.30E-08	2.21E-02	1.31E-08	2.22E-02	4.34E-08	7.37E-02	4.37E-08	7.42E-02	6.51E-08	1.11E-01	6.57E-08	1.12E-01
7	2.51E+05	1.53E-06	3.84E-01	1.53E-06	3.84E-01	1.53E-06	3.84E-01	1.53E-06	3.84E-01	1.53E-06	3.84E-01	1.53E-06	3.84E-01
8	1.42E+06	4.12E-07	5.84E-01	4.12E-07	5.84E-01	4.12E-07	5.84E-01	4.12E-07	5.84E-01	4.12E-07	5.84E-01	4.12E-07	5.84E-01
Total		6.06E-06	1.71E+00	6.06E-06	1.71E+00	6.06E-06	1.78E+00	6.06E-06	1.78E+00	6.06E-06	1.83E+00	6.06E-06	1.83E+00
CCFP		52.47%		52.48%		52.98%		52.98%		53.33%		53.34%	
ΔCCFP				0.00%		0.50%		0.51%		0.86%		0.87%	
ΔLERF				5.05E-11		3.04E-08		3.06E-08		5.20E-08		5.27E-08	
Change in Dose Risk					8.50E-05		7.10E-02		7.15E-02		1.22E-01		1.23E-01
Percent Change in Dose Risk					0.005%		4.16%		4.19%		7.14%		7.21%

The ΔCCFP, ΔLERF, Change in Dose Risk and Percent Change in Dose Risk are all calculated with respect to the Base Case Without Corrosion.

Corrosion Sensitivity Study Results:

- The ΔCCFP for both the 10-year and 15-year interval still fall into the definition of "a small increase" as defined in EPRI 1018243.
- The ΔLERF for both the 10-year and 15-year interval still fall into the category of "a very small change in risk" per the Regulatory Guide 1.174 definition of an increase in LERF of less than 1.0E-07 per reactor year.
- The change in dose risk for both the 10-year and 15-year interval still meet the definition of "a very small population dose" as defined in EPRI 1018243.

Evaluation of the Sensitivity to Corrosion Parameter Assumptions

Table 6-3 indicates that including the corrosion effects using the assumptions described in Table 6-1 insignificantly adds to the risk of the ILRT extension.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters used in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood for the dome and cylinder was adjusted from 10% to 15% and 5%. The results are presented in Table 6-4.

The first year failure rate used for doubling the failure rate every 2 years and every 10 years was assumed to be the same as for the 5 year base case, namely 2.1E-03/yr for the Cylinder and Dome and 5.1E-04 for the Basemat. The doubling rate is determined according to the following formula: Doubling Rate (%/yr) = 100 * ($2^{1/x}$ - 1), where "x" is the number of years between doublings.

Table 6-4: Steel Liner Corrosion Sensitivity Cases

Age (Table 6-1, Step 3)	**Containment Breach (Table 6-1, Step 4)**	**Inspection Detection Failure (Table 6-1, Step 5)**	**Increase in Class 3b Frequency (LERF) for ILRT Extended from 3 to 15 years (per yr)**	
		(Cylinder only, Basemat is always 100%)	**Total Increase**	**Increase Due to Corrosion**
Base Case, Doubles Every 5 Years (14.9%/yr)	Base Case, 1% Cylinder, 0.1% Basemat	Base Case, 10%	5.27E-08	6.34E-10
Doubles Every 2 Years (41.4%/yr)	Base	Base	5.83E-08	6.26E-09
Doubles Every 10 Years (7.2%/yr)	Base	Base	5.24E-08	3.24E-10
Base	Base	15%	5.29E-08	8.88E-10
Base	Base	5%	5.24E-08	3.81E-10
Base	10% Cylinder, 1% Basemat	Base	5.84E-08	6.34E-09
Base	0.1% Cylinder, 0.01% Basemat	Base	5.21E-08	6.34E-11
Lower Bound, Doubles Every 10 Years	0.1% Cylinder, 0.01% Basemat	5%	5.21E-08	1.94E-11
Upper Bound, Doubles Every 2 Years	10% Cylinder, 1% Basemat	15%	1.40E-07	8.76E-08

In every case, the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield an increase in LERF due to corrosion of only 8.76E-08 /yr. The results indicate that even with very conservative assumptions, the conclusions of the base analysis remain valid.

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CONCLUSIONS

The assessment of the plant risk associated with extending the Type A ILRT frequency from three in ten years to one in fifteen years concludes that:

- The increase in LERF, including the potential effect of liner corrosion is 5.27E-08/yr. Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6} /yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.
- The change in total population dose risk, including the potential effect of liner corrosion, is 0.123 person-rem/yr. EPRI 1018243 [2] states that a very small population dose is defined as an increase of ≤1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The change in population dose is classified as "very small." This risk impact, when compared to other severe accident risks, is negligible.
- The increase in the conditional containment failure probability, including the potential effect of liner corrosion, is 0.87%. EPRI 1018243 [2] states that increases in CCFP of ≤1.5 percentage points are small and, therefore, this is classified as "a small increase."

The overall conclusion is that permanently increasing the ILRT interval to once every 15 years is acceptable since it represents a very small increase in the overall South Texas Project risk profile.

REFERENCES

1) NEI 94-01, Revision 3A, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J.*
2) EPRI 1018243, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, Revision 2-A of 1009325*
3) NRC RG 1.200, Revision 2, *An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities.*
4) NRC RG 1.174, Revision 2, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.*
5) NRC ADAMS ML020920100, *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension,* Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.
6) NUREG-1493, *Performance-Based Containment Leak Test Program.*
7) EPRI TR-104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals.*
8) STPEGS UFSAR Section 13.7, *STP Exemption from Special Treatment Requirements.*
9) NOC-AE-01001115, *South Texas Project Units 1 And 2 Docket Nos. STN 50-498, STN 50-499 Proposed Amendment to Technical Specification 6.8.3.j for a Change in 10CFR50, Appendix J, Integrated Leakage Rate Test Interval.*

10) NOC-AE-02001323, *South Texas Project Units 1 And 2 Docket Nos. STN 50-498, STN 50-499 Response to Request for Additional Information - South Texas Project Containment Integrated Leakage Rate Test Interval Extension.*

11) NOC-AE-02001275, *South Texas Project Units 1 And 2 Docket Nos. STN 50-498, STN 50-499 Addendum To Proposed Amendment To Technical Specifications 6.8.3.j for a Change in 10CFR50, Appendix J, Integrated Leakage Rate Test Interval.*

12) NOC-AE-10002607, *Letter from G. T. Powell, STP, to NRC Document Control Desk, "License Renewal Application", dated October 25, 2010, Appendix E, Attachment F*

13) STP_RV72 PRA, STI 33651684, *South Texas Project Level I and II Approved PRA*, 1/31/13.

14) STP_RV6 PRA, STI 32746637 *South Texas Project Level I and II Approved PRA*, 7/30/09.

15) STP_RV7 Level 2 Analysis Containment Event Tree Notebook, STI 33949540, Reviewed 6/26/12.

16) STP_RV7 Level 2 Accident Progression Notebook, STI 33949538, Reviewed 6/26/12.

17) *Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

18) NUMARC, ILRT Survey Data, February 18, 1994.

19) NEI ILRT Survey, 2001

20) License Amendments No. 188 and No. 175 for Units 1 and 2, respectively, dated October 31, 2008, providing for a revision to the Technical Specifications that relocates the frequencies of most surveillance tests from the T.S. to the SFCP (ST-AE-NOC-08001822, STI 32394178).

21) Amendment No. 179 & No. 166, dated July 13, 2007. This application approved a broad-scope risk-informed technical specifications or Risk Managed Technical Specifications.

22) 0PSP11-IL-0007, Rev. 7, Reactor Containment Building Integrate Leakage Rate Test

23) NOC-AE-07002112, *STP Units 1 & 2 Response to NRC Requests for Additional Information on STPNOC Proposed Risk Managed Technical Specifications (TAC Nos. MD 2341 & MD 2342)*

24) AE-NOC-07001652, NRC ADAMS ML071780186, *South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Broad-Scope Risk-Informed Technical Specifications Amendments (TAC Nos. MD2341 And MD2342)*

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COMPUTER INPUT/OUTPUT

Model: STP RV72
Master Frequency File: MFFREV7P
Bin Totals for All Initiators
2:30 PM 1/7/2015
Page 1

Bin	Frequency (Quantified)	Frequency (Saved)	Bin Description
MELT	6.0624E-006		
INTACT2	2.5947E-006		
R05SU	8.1438E-007		
R15U	7.1821E-007		
R13U	3.6838E-007		
ISGTR	3.6658E-007		
R11U	2.1096E-007		
INTACT1	1.6754E-007		
R07SU	1.3955E-007		
R09U	8.2210E-008		
R05SLU	8.0039E-008		
CICV	4.2113E-008		
R06SU	3.3810E-008		
R13	2.7908E-008		
BYPASS	2.6674E-008		
R14	2.4790E-008		
R06S	2.1117E-008		
R07SLU	1.3429E-008		
R08SU	9.8630E-009		
R09	8.9014E-009		
R08S	7.5588E-009		
R14U	6.4208E-009		
R11	5.9951E-009		
R05S	4.0261E-009		
R06SLU	3.8446E-009		
R01U	3.0025E-009		
R03U	1.5836E-009		
R06SL	5.0012E-010		
R16U	4.5763E-010		

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Model: STP_RV72
Master Frequency File: MFFREV7P
Bin Totals for All Initiators
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Bin	Frequency (Quantified)	Frequency (Saved)	Bin Description
R10	2.8738E-010		
R01	2.7199E-010		
R08SL	2.1014E-010		
R08SLU	1.9491E-010		
R03	1.6193E-010		
R05SL	1.1636E-010		
R16	1.0065E-010		
R15	6.4782E-011		
R05U	2.7635E-011		
R07S	2.6044E-011		
VSEQ	5.2081E-012		
R07U	3.2806E-012		
R04	0.0000E+000		
R02U	0.0000E+000		
R02	0.0000E+000		
R05	0.0000E+000		
R07L	0.0000E+000		
R04U	0.0000E+000		
OTHER	0.0000E+000		
R07SL	0.0000E+000		
R12U	0.0000E+000		
R12	0.0000E+000		
R10U	0.0000E+000		
R08U	0.0000E+000		
R08LU	0.0000E+000		
R06U	0.0000E+000		
R08	0.0000E+000		
R05L	0.0000E+000		
R07LU	0.0000E+000		

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Model: STP_RV72
Master Frequency File: MFFREV7P
Bin Totals for All Initiators

2:30 PM 1/7/2015
Page 3

Bin	Frequency (Quantified)	Frequency (Saved)	Bin Description
R07	0.0000E+000		
R06LU	0.0000E+000		
R06L	0.0000E+000		
R06	0.0000E+000		
R05LU	0.0000E+000		
R08L	0.0000E+000		
	1.1848E-005		

ADDENDUM 1- ILRT Risk Assessment Resource Documents

The following industry studies for containment leakage risk assessment are briefly described and summarized here:

1. NUREG/CR-3539
2. NUREG/CR-4220
3. NUREG-1273
4. NUREG/CR-4330
5. EPRI TR-105189
6. NUREG-1493
7. EPRI TR-104285
8. NUREG-1150 and NUREG/CR-4551
9. NEI Interim Guidance
10. Calvert Cliffs liner corrosion analysis
11. EPRI Report No. 1009325, Revision 2, Appendix H

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for the South Texas Project. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent 15-year extension of the ILRT interval.

NUREG/CR-3539

Oak Ridge National Laboratory documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage.

NUREG-1273

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies: "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The conclusion from the study is that a small but measurable safety benefit is realized from extending the test intervals.

NUREG-1493

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies: Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.

Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals. EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded: "... the proposed CLRT [containment leak rate tests] frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.04 person-rem per year..."

NUREG-1150 and NUREG/CR 4551

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent any plant. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

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NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals

The guidance provided in this document builds on the EPRI risk impact assessment methodology and the NRC performance-based containment leakage test program, and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms were factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner.

Licensees may consider approved LARs for one-time extensions involving containment types similar to their facility. This assessment has addressed the plant-specific differences from the Calvert Cliffs design, and how the Calvert Cliffs methodology was adapted to address the specific design features. In the case where no similar analyses has been performed the licensee will use judgment based the available analyses and plant specific features to perform the analysis.

EPRI Report No. 1009325, Revision 2-A, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*

This report provides a generally applicable assessment of the risk involved in extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant-specific supplemental risk impact assessments and builds on the previous EPRI risk impact assessment methodology and the NRC performance-based containment leakage test program, and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

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PRA Analyses/Assessments			
Form 4	Reviewer Checklist and Comment Sheet		

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Yes

☒ The analysis/assessment contains an accurate and complete Purpose and Scope per Addendum 2 step 1.4.1.

☒ The analysis/assessment correctly describes the Method of Analysis per Addendum 2 step 1.4.2.

☒ The analysis/assessment correctly identifies computer inputs/outputs and these have been verified correct per Addendum 2 step 1.4.3.

☒ The analysis/assessment clearly identifies assumptions per Addendum 2 step 1.4.4, and these assumptions have been validated.

☒ The analysis/assessment clearly and correctly presents the results per Addendum 2 step 1.4.5.

☒ The analysis/assessment identifies the correct references per Addendum 2 step 1.4.6.

☒ The analysis/assessment results are <u>separately reproduced</u> and correct per Addendum 2 step 2.1.

☒ Software used by this analysis/assessment is approved as required by the STP Software Quality Assurance Program, 0PGP07-ZA-0014.

(___ Checkmark here for N/A if the assessment was qualitative)

☒ The Preparer and Reviewer are qualified to Engineering Support Program (ESP) Certification 9287, "Perform Risk-Based Safety Assess/Evaluations" as shown in the Qual King database.

This form, when completed, shall be retained for the life of plant.

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	0PGP05-ZE-0001	Rev. 3	
PRA Analyses/Assessments			
Form 4	Reviewer Checklist and Comment Sheet		

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Reviewer's analysis/assessment summary reproducing assessment results (by alternate calculations or separately reproducing the analysis):

Separately reviewed all inputs to assessment and came to same conclusions.

The review included verifying that this assessment was performed in accordance with the guidance provided in EPRI 1018243. Beaver Valley's LAR submittal was also referenced.

All calculations specific to STP were separately reproduced. Any issues found were resolved.

- STP_RV72 was used to generate Level 2 Release Bins as shown in Table 1-2. A scaling factor was determined and used to adjust frequencies.
- Table 4-2 in the Level 2 CET Notebook (Release Category Assignment Matrix) was used to review the classifications of STP release categories into the corresponding EPRI release BINS grouping.
- The EPRI sub-classes were reproduced independently in Excel.
- The population dose for represented accident classes used were verified against the SAMA analysis (STI 32889793)
- The dose risk was calculated independently using Excel.
- The risk impact was evaluated in terms of population dose rate and percentile change in population does rate using Excel. This was independently verified.
- The risk impact in terms of change in LERF and the change in CCFP was independently reproduced using Excel.
- The liner corrosion analysis and sensitivity was independently reproduced in Excel, including Beaver Valley analysis results for comparison purposes.

This form, when completed, shall be retained for the life of plant.