FPL.

August 26, 2014

L-2014-230
10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Re: St. Lucie Nuclear Plant
Docket Nos. 50-335 and 50-389
License Amendment Request
Application for Technical Specifications Change to Permanently Extend the Integrated Leak Rate Test (ILRT) Frequency to 15 Years

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) is submitting a request for amendments to the Technical Specifications (TS) for St. Lucie Unit 1 and Unit 2.

The proposed amendments would modify TS requirements for containment leakage rate testing by extending the 10-year ILRT (Type A test) interval to 15 years on a permanent basis in accordance with the provisions of NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J."

Approval of the proposed amendment is requested within one year of the submittal date. Please note that for Unit 1 the next 10-year ILRT is due to be performed no later than December 8, 2015, as required by TS 6.8.4.h, "Containment Leakage Rate Testing." If an amendment is not received prior to then, Unit 1 must be shut down to perform the ILRT. Once approved, the amendments will be implemented within 60 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Florida Official.

This license amendment request has been reviewed by the St. Lucie Onsite Review Group.

This license amendment request makes no new commitments or changes to any other existing commitments.

If you should have any questions regarding this submittal, please contact Eric Katzman, Licensing Manager, at (772) 467-7734.

A017
NRR

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

Executed on August 26, 2014

Sincerely,

(ACTING)
Joseph Jensen
Site Vice President
St. Lucie Nuclear Plant

Enclosure

cc: USNRC Regional Administrator, Region II
USNRC Project Manger, St. Lucie Nuclear Plant
USNRC Senior Resident Inspector, St. Lucie Nuclear Plant
Ms. Cynthia Becker, Chief – Florida Bureau of Radiation Control

Enclosure

Subject: License Amendment Request to Extend Integrated Leak Rate Test Frequency to 15 Years

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1.0 SUMMARY DESCRIPTION

Florida Power & Light Company (FPL) requests to revise Technical Specification (TS) 6.8.4.h, "Containment Leakage Rate Testing Program," for St. Lucie Unit 1 and Unit 2. The proposed change would allow for the extension of the St. Lucie 10-year integrated leak rate test (ILRT or Type A test) interval to 15 years on a permanent basis in accordance with the provisions of NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J."

The technical basis for the proposed license amendment utilizes risk-informed analysis augmented with non-risk related considerations. A risk impact evaluation performed by Westinghouse Electric Company (WEC) concluded that the increases in large early release frequency (LERF) are within the limits set forth by the applicable guidance contained in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis"; NUREG-1493, "Performance-Based Containment Leak-Test Program"; and Electric Power Research Institute (EPRI) Technical Report TR-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The technical basis provides a determination that the proposed amendment does not involve a significant hazards consideration as described in 10 CFR 50.92.

In accordance with the guidance of NEI 94-01, Revision 3-A, St. Lucie proposes to extend the maximum surveillance interval for the ILRT to no longer than 15 years from the last ILRT based on satisfactory performance history. The current interval is no longer than 10 years and would require that the next ILRT for Unit 1 be performed during the Spring 2015 refueling outage and for Unit 2 during the Spring 2017 refueling outage. The proposed change would allow Unit 1 and Unit 2 ILRTs to be performed in 2020 and 2022, respectively. This will reduce the number of ILRTs performed over the licensed period of operation for both units resulting in significant savings in radiation exposure to personnel, cost, and critical path time during refueling outages.

2.0 DETAILED DESCRIPTION

<u>Unit 1</u>

Unit 1 TS 6.8.4.h, "Containment Leakage Rate Testing Program," describes this program in part as:
A program 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 is in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," as modified by the following exception(s):

a) Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by RG 1.163) will be used for Type A testing.
b) The first Type A test performed after the May 1993 Type A test shall be no later than May 2008.

The proposed amendment would replace the reference to RG 1.163 with reference to the latest revision of the applicable NEI standard addressing a performance-based program and surveillance intervals. The existing exceptions would be deleted, as they are no longer necessary or applicable, and replaced with the next ILRT (Type A test) surveillance due date. The cited section of TS 6.8.4.h would be revised to state:

> A program 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 is in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J," except that the next Type A test performed after the December 8, 2005 Type A test shall be performed no later than December 8, 2020.

Unit 2

Unit 2 TS 6.8.4.h, "Containment Leakage Rate Testing Program," describes this program in part as:
A program 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 is in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," as modified by Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by RG 1.163) which will be used for type A testing.

The proposed amendment would replace the reference to RG 1.163 with reference to the latest revision of the applicable NEI standard addressing a performance-based program and surveillance intervals. The existing exception(s) would be deleted, as they are no longer necessary or applicable, and replaced with specific reference to the next ILRT (Type A test) surveillance due date. The cited section of TS 6.8.4.h would be revised to state:

> A program 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 is in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 18, 2007 Type A test shall be performed no later than December 18, 2022.

In addition, an editorial correction in a subsequent paragraph within the same TS section is proposed for clarity.

TS markups for the proposed change are provided in Attachments 1 and 2, and retyped TS pages reflecting the proposed change are presented in Attachments 3 and 4 for Units 1 and 2, respectively.

2.1 Background

The testing requirements of 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," provide assurance that leakage through the containment, including systems and components that penetrate containment, does not exceed the allowable leakage values as specified in Technical Specifications, and that periodic surveillance of the primary containment penetrations and isolation valves is performed so that proper maintenance and repairs are made over the service life of the containment, and systems and components penetrating primary containment. The limitation on containment leakage, when tested at design accident pressure, provides assurance that the containment and its appurtenances would perform their design function. Appendix J defines 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 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 containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those parts of the containment not covered by Type B and C testing.

10 CFR 50 Appendix J was revised effective October 26, 1995 to allow use of Option B, "Performance-Based Requirements." Option B requires that the test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. The performance-based test intervals are based on consideration of the structure or component operational history and the resulting risk from its failure. The revision was focused on improving the body of regulations by eliminating prescriptive requirements that are marginal to safety and providing licensees greater flexibility for cost effective implementation methods to meet regulatory safety objectives.

RG 1.163, "Performance-Based Containment Leak Test Program," was issued in September 1995 as an acceptable method to the NRC for implementing Option B. RG 1.163 states that NEI 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," prepared by the Nuclear Energy Institute (NEI), provides methods acceptable to the NRC staff for complying with the provisions of Option B in Appendix J to 10 CFR 50, subject to certain exceptions. NEI 94-01, Revision 0 allows licensees to perform the ILRT 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 leakage rate was less than the allowable leakage rate. The change was based on NUREG-1493 and EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," both of which established that the risk increase associated with extending the ILRT surveillance interval was very small.

In 1996, FPL submitted TS change requests for Unit 1 and Unit 2 to adopt Option B of 10 CFR 50, Appendix J. NRC approval was granted by letter dated February 10, 1997 in which these requests were approved as Amendments 149 and 88, respectively. Based on the established performance history, Unit 1 and Unit 2 were placed on a 10-year ILRT surveillance interval.

In 2002, FPL submitted TS change requests for Unit 1 and Unit 2 for a one-time ILRT interval extension to 15 years. This was approved by NRC letter dated April 14, 2003, as Amendments 187 and 130, respectively. The Unit 1 interval became due no later than May 2008; the Unit 2 interval became due no later than June 2007.

In 2005, FPL submitted a TS change request for Unit 2 to obtain a one-time extension of the 15-year interval in order to reach the SL2-17 refueling outage which was scheduled in late 2007. This was done to eliminate the need to perform ILRTs in both the SL2-16 and SL2-17 outages since the containment was being cut open for reactor vessel head and steam generator replacement during the SL2-17 outage. This was approved by NRC letter dated December 30, 2005 as Amendment 140. The Unit 2 ILRT became due no later than SL2-17 refueling outage, which effectively added 6 months to the existing interval.

In 2007, NEI submitted NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, "Risk impact Assessment of Extended Integrated Leak Rate Testing Intervals," to the NRC staff for review.

In the safety evaluation (SE) issued by the NRC dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2 describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.1 of the SE. The accepted version of NEI 94-01 was subsequently issued as Revision 2-A dated October 2008. The NRC noted in the SE that NEI 94-01, Revision 2 incorporates the regulatory positions stated in RG 1.163.

EPRI TR-1009325, Revision 2 provides a risk impact assessment for optimized ILRT intervals of up to 15 years, using current industry performance data and risk-informed guidance, primarily Revision 1 of RG 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The NRC's final SE issued by letter dated June 25, 2008, documents the evaluation and acceptance of EPRI TR-1009325, Revision 2, subject to the specific limitations and conditions listed in Section 4.2 of the SE. An accepted version of EPRI TR-1009325 was subsequently issued as Revision 2-A (also identified as TR-1018243) dated October 2008.

In 2012, the NRC determined NEI 94-01, Revision 3-A was acceptable for referencing in licensing applications subject to the limitations and conditions of Section 4.0 of the accompanying SE. These limitations and conditions stemmed from the inclusion of guidance for extending the Type C (local leak rate test or LLRT) interval to 75 months.

As described in an NRC Letter to NEI dated August 20, 2013, "Request Revision to Topical Report NEI 94-01 Revision 3-A," Revision 3-A did not include the six limitations and conditions in the NRC's June 25, 2008 SE approving NEI 94-01, Revision 2. Although the six limitations and conditions were not included in NEI 94-01, Revision 3-A, they apply to a licensee's request to use NEI 94-01, Revision 3-A to extend the ILRT interval.

3.0 TECHNICAL EVALUATION

The primary containment provides a low-leakage barrier against the uncontrolled release of radioactivity into the environment following a design basis accident. The requirements set forth in 10 CFR 50 Appendix J provide assurance that the leakage from the primary containment does not exceed the allowable values specified in TSs. Option B of 10 CFR 50 Appendix J requires that the test intervals for Type A, Type B, and Type C testing be determined using a performance-based approach. Currently the St. Lucie Containment Leakage Rate Testing Program (TS 6.8.4.h) is based on RG 1.163 which endorses NEI 94-01, Revision 0. This license amendment proposes to update the St. Lucie program by implementing the guidance of NEI 94-01, Revision 3-A.

The following table addresses each of the six limitations and conditions associated with NRC approval of Revision 2 and the two limitations and conditions associated with NEI 94-01, Revision 3-A.

Limitation/Condition Section 4.1 of SE for Revision 2	**FPL Response**
1. 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.	The St. Lucie Containment Leakage Rate Testing Program (ADM-68.01) utilizes the definition found in section 5.0 of NEI 94-01, Revision 3-A for calculating the Type A test leakage rate. Revision 3-A contains the same definition as Revision 2.
2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests.	A schedule of containment inspections is provided in Section 3.3 of this submittal.
3. The licensee addresses the areas of containment structure potentially subjected to degradation.	General visual examination of accessible interior and exterior surfaces of the containment system for structural problems is conducted in accordance with the St. Lucie In-Service Inspection Plan which implements the requirements of ASME Section XI, Subsection IWE, as required by 10 CFR 50.55a(g). Areas selected for augmented inspection in accordance with IWE-1240 are discussed in Section 3.4 of this submittal.
4. The licensee addresses any test and inspections performed following major modifications to the containment structure as applicable.	The welded construction hatch on Unit 1 was removed for replacement of the reactor vessel head and reactor coolant system pressurizer in SL1-20 (Fall 2005). A successful Type A test was performed after restoration of the construction hatch. The welded construction hatch on Unit 2 was removed for replacement of the reactor vessel head and steam generators in SL2-17 (Fall 2007). A successful Type A test was subsequently performed after restoration of the construction hatch.
5. The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provisions 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.	FPL acknowledges and accepts the NRC staff position as communicated to the nuclear industry in Regulatory Issue Summary (RIS) 2008-27 dated December 8, 2008.
6. For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2 and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	This is not applicable to St. Lucie Unit 1 and Unit 2. Neither unit is licensed pursuant to 10 CFR Part 52.

Limitation/Condition Section 4.0 of SE for Revision 3-A	FPL Response
1. NEI TR 94-01, Revision 3 is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs to be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84 months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI TR 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g., BWR MSIVs), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months. This is Topical Report Condition 1.	The St. Lucie Containment Leakage Rate Testing Program does not allow for Type C (LLRT) interval extension beyond 60 months. NEI 94-01, Revision 3-A,will be cited in TS 6.8.4.h for Units 1 and 2 as the governing guidance, and a procedure change request has been approved for update of ADM-68.01, "Containment Leakage Rate Testing Program," to reflect the requirements for an interval extension upon approval of this license amendment request. Consequently, FPL would provide the information, as requested in this condition, if a program change is ever made and an interval extension up to 75 months is utilized.
2. The basis for extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time. The containment leakage condition monitoring regime involved a portion of the penetration being tested each refueling outage, nearly all LLRTs being performed during plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leakage rates for the just tested penetrations are summed with the as-left minimum pathway leakage rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves which, in the aggregate, will show increasing leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate	The St. Lucie Containment Leakage Rate Testing Program does not allow for Type C (LLRT) interval extension beyond 60 months. NEI 94-01, Revision 3-A,will be cited in TS 6.8.4.h for Units 1 and 2 as the governing guidance, and a procedure change request has been approved for update of ADM-68.01, "Containment Leakage Rate Testing Program," to reflect the requirements for an interval extension upon approval of this license amendment request. Consequently, FPL would provide the information, as requested in this condition, if a program change is ever made and an interval extension up to 75 months is utilized.

Limitation/Condition Section 4.0 of SE for Revision 3-A	FPL Response
2. (continued) maintenance may extend this increasing leakage potential. Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plant consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1. When routinely scheduling any LLRT valve interval beyond 60 months and up to 75 months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating the LLRT totals calculated represent the actual leakage potential of the penetrations. This is Topical Report Condition 2.	

3.1 Previous Type A Test (ILRT) Results

A total of seven ILRTs, including the pre-operational test, have been performed on Unit 1, all with satisfactory results. A total of five ILRTs, including the pre-operational test, have been performed on Unit 2, all with satisfactory results. As required by NEI 94-01, Revision 3-A Section 9.1.2, further extensions in test intervals are based upon two consecutive successful periodic Type A tests and the requirements stated in Section 9.2.3 of this guideline. There has been substantial margin to the performance limit as described in TS of La equal to 0.5% wt/day.

To provide context for the results, the performance leak rate definition from NEI 94-01, Revision 3-A Section 5.0 follows. The performance leakage rate is calculated as the sum of the Type A upper confidence limit (UCL) and as-left minimum pathway leakage rate (MNPLR) rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position (i.e., drained and vented to containment atmosphere) prior to performing the Type A test. In addition, leakage pathways that are isolated during performance of the test because of excessive leakage must be factored into the performance determination. The performance criterion for Type A tests is a performance leak rate of less than 1.0 La.

The following is a description of the results from the two prior Unit 1 ILRTs.

The December 2005 periodic Type A test using the ANSI Mass Point method calculated at the 95% UCL resulted in a leakage rate of 0.2033% wt/day. The performance leak rate corresponding to the definition in NEI 94-01 was 0.2102% wt/day with corrections.

The May 1993 periodic Type A test, formally performed using the BN-TOP-1, calculated at the 95% UCL resulted in a leakage rate of 0.293% wt/day. The ILRT Report notes that the computer calculated corresponding ANSI 56.8 Mass Point 95% UCL leakage rate was 0.133% wt/day. The reported BN-TOP-1 value is higher primarily due to conservatisms of the BN-TOP-1 calculation. Using the preferred ANS method results in a performance leak rate corresponding to the definition in NEI 94-01 of 0.133% wt/day with corrections.

The following is a description of the results from the two prior Unit 2 ILRTs.

The December 2007 periodic Type A test using the ANSI Mass Point method calculated at the 95% UCL resulted in leakage rate of 0.1911% wt/day. The performance leak rate corresponding to the definition in NEI 94-01 was 0.1930% wt/day with corrections.

The June 1992 periodic Type A test, formally performed using the BN-TOP-1, calculated at the 95% UCL resulted in a leakage rate of 0.052% wt/day. The performance leak rate corresponding to the definition in NEI 94-01 was 0.052% wt/day with corrections.

These results show that there is a considerable margin compared to the maximum allowable leakage rate of 0.5% wt/day. There are no containment modifications planned for Unit 1 prior to 2020 or for Unit 2 prior to 2022. Any unplanned modifications to the containment prior to the next scheduled Type A test would be subject to the special testing requirements (Section IV.A) of 10 CFR 50 Appendix J, "Containment Modification."

There have been no pressure or temperature excursions in the containment which could have adversely affected containment integrity. There is no anticipated addition or removal of plant hardware within containment which could affect leak-tightness.

3.2 Type B and Type C Testing (LLRT) Program

The St. Lucie (PSL) Appendix J LLRT program requires testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B and TS 6.8.4.h. The program is delineated in St. Lucie procedure ADM-68.01, "Containment Leakage Rate Testing Program."

When a component fails to meet its respective administrative limit, it is evaluated using the St. Lucie corrective action process (CAP) and placed on a test frequency of each outage (or 24 months). To permit operation at Extended Power Uprate (EPU) conditions, the NRC issued Amendment 213 (ML12156A208), which raised the Unit 1 Pa value from 39.6 psig to 42.8 psig, and Amendment 163 (ML12235A463), which raised the Unit 2 Pa value from 41.8 psig to 43.5 psig. Currently all components within the scope of the LLRT program are being tested at a test frequency of each refueling outage. This was commenced prior to approval of EPU operation for both units to ensure compliance with the requirements for performing LLRT at or above Pa pressure and to conservatively establish a new performance history even though the new Pa values are not significantly higher. Currently St. Lucie has not begun the reevaluation process for extended test intervals.

Attachment 6 contains listings of respective LLRT failures dating back to the period of the last ILRT for each unit. When a component fails to meet the administrative limit it is repaired and retested prior to returning it to operable status for a mode in which it is required. Note that as-left failures do not represent a final as-left condition and are included only to provide a complete representation of instances in which components did not meet administrative limits during LLRT.

As discussed in NUREG-1493 and NEI 94-01, Revision 3-A, Type B and Type C tests can identify the vast majority of all containment leakage paths. This license amendment request adopts the guidance in NEI 94-01, Revision 3-A in place of NEI 94-01, Revision 0, but otherwise does not affect the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that containment leakage rates are maintained well within limits.

3.3 Supplemental Inspections

The containment vessel, including all its penetrations, is a low leakage steel shell designed to withstand a postulated design basis accident (DBA) and to confine the radioactive materials that could be released by accidental loss of integrity of the reactor coolant pressure boundary. The containment vessel is a right circular cylinder (approximately 2 in. thick) with a hemispherical dome (approximately 1 in. thick) and ellipsoidal bottom (approximately 2 in. thick). The containment vessel is equipped with a dome inspection walkway, access ladder, and circular crane girder with a crane rail attached to the shell of the vessel. The containment vessel is enclosed by the reinforced concrete Shield Building. An annular space is provided between the walls and domes of the containment vessel and the Shield Building in order to permit construction operations, in-service inspection, and to filter any leakage from containment during a loss of coolant accident (LOCA) to minimize site doses.

RG 1.163 Regulatory Position C.3 specifies that visual examinations of accessible interior and exterior surfaces of the containment system are to be performed to evaluate for structural problems. These examinations are to be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the next Type A test has been extended to 10 years. NEI 94-01, Rev 3-A Section 9.2.3.2, "Supplemental Inspection Requirements," specifies that a general visual examination of the accessible interior and exterior surfaces of containment be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years. St. Lucie has established programs for performing visual examinations of the accessible surfaces of the containment for detection of structural deterioration.

A general visual inspection of the accessible interior and exterior containment surfaces is performed prior to the Type A test (ILRT) and once per period in accordance with the IWE Program schedule, 2nd-Interval-IWE-PSL-1-Schedule, and 2nd-Interval-IWE-PSL-2-Schedule. Each 10-year interval is divided into 3 periods of approximately equal duration ensuring that the requisite frequency is maintained.

Prior to the start of the 2nd ten-year interval in Sept 2008, the IWE general visual inspection was only performed on the interior surface. Subsequently, the IWE general visual inspection was revised to include the exterior of the steel containment vessel. Prior to implementation of the IWE inspections, the St. Lucie Containment Leakage Rate Testing Program required a general visual inspection of the accessible interior and exterior containment surfaces at a frequency of two times in 10 years and prior to an ILRT. In 2005 the Containment Leakage Rate Testing Program inspection requirement was changed to combine the inspection of the interior of the steel containment vessel with the IWE inspection. Subsequently, the inspection of the exterior steel containment vessel was also combined.

The general visual inspection of the interior and exterior containment surfaces has been performed twice on each unit since the beginning of the 2nd interval in 2008. Also, during the 2nd interval, 100 percent of the moisture barrier, both interior and exterior, have been inspected per period, as required by ASME Section XI, 2001 Edition with 2003 Addenda, Subsection IWE.

The following provides an approximate schedule for the general visual inspection of the accessible containment vessel surfaces assuming the Type A test frequency is extended to 15 years.

PSL-1
Table 4.3-1

Refueling Outage	Year	ILRT	General Visual Examination of Accessible Internal Surface	General Visual Examination of Accessible External Surface
SL1-20	F2005	X	X	X
SL1-21	S2007			
SL1-22	F2008		X	X
SL1-23	S2010			
SL1-24	W2011		X	X
SL1-25	F2013			
SL1-26	S2015			
SL1-27	F2016		X	
SL1-28	S2018			X
SL1-29	F2019	X	X	X
SL1-20	S2021			

PSL-2
Table 4.3-2

Refueling Outage	Year	ILRT	General Visual Examination of Accessible Internal Surface	General Visual Examination of Accessible External Surface
SL2-17	F2007	X	X	X
SL2-18	S2009			
SL2-19	S2011		X	X
SL2-20	F2012		X	
SL2-21	S2014			X
SL2-22	F2015		X	
SL2-23	S2017			X
SL2-24	F2018			
SL2-25	S2020			
SL2-26	F2021	X	X	X
SL2-27	S2023			

The accessible interior and exterior surfaces of the concrete Shield Building continue to be inspected at a similar frequency in accordance with the Containment Leakage Rate Testing Program as required by St. Lucie Technical Specifications. This inspection is performed independently from the IWE inspections and provides another opportunity to observe any potential deficiencies of the exterior surfaces of the steel containment vessel.

Accessible Service Level 1 coatings inside containment are inspected each outage in accordance with St. Lucie coatings controls procedures. The containment vessel and coated appurtenances are included in the scope of this activity. The primary purpose of these inspections is to minimize the potential for clogging of the containment sump strainers; however, these inspections also serve to ensure that the containment surface coatings are maintained, and by maintaining these coatings, serves to protect the containment vessel from deterioration.

Inspection results indicate that no significant corrosion effects have been experienced involving the containment vessel and penetrations. These results demonstrate that the program continues to be an acceptable means to ensure that the containment is capable of maintaining its design basis integrity function.

3.4 Deficiencies Identified

Deficiencies observed during the IWE and other inspection processes include localized instances of coatings deficiencies, disbondment of moisture barrier seals, and mild surface corrosion. All deficiencies are addressed using the CAP to remediate the identified condition. There have been no significant defects noted from the results of these inspections.

In accordance with the criteria established in IWE-1240 the following areas have been identified for augmented inspection applicable to both St. Lucie Unit 1 and Unit 2. These areas are included in 2nd-Interval-IWE-PSL-1-Schedule and 2nd-Interval-IWE-PSL-2-Schedule.

(1) The steel containment vessel (annulus side) shell plate extending from the north side of the transfer canal to a location approximately 10 feet west of the maintenance hatch. This area extends vertically from the floor-to-shell interface up to the elevation of the bottom of the maintenance hatch.

(2) The steel containment vessel (annulus side) shell plate in the area of the mechanical penetrations in IWE Zones 4 and 5 (extending from azimuth 133 to azimuth 190 - approximate). This area extends vertically from the floor-to-shell interface up to elevations accessible from the 23' elevation (i.e., up to 29' elevation - approximate). Inspection shall include corresponding penetrations accessible from the 23' elevation (i.e., up to 29' elevation - approximate).

(3) The steel containment vessel (containment side) shell plate extending 10 feet on either side of the maintenance hatch. This area extends vertically from the floor-to-shell interface up to the elevation of the bottom of the maintenance hatch.

(4) The steel containment vessel (containment side) shell plate in the area of the mechanical penetrations in IWE Zones 4 and 5 (extending from azimuth 133 to azimuth 190 - approximate). This area extends vertically from the floor-to-shell interface up to elevations accessible from the 23' elevation (i.e., up to 29' elevation - approximate). Inspection shall include corresponding penetrations accessible from the 23' elevation (i.e., up to 29' elevation - approximate).

Prior to implementation of the IWE inspection program there were two areas of interest identified, as documented in a prior FPL licensing submittal: FPL letter L-2002-143, "Proposed License Amendments - Risk-Informed One Time Increase in Integrated Leak Rate Test Surveillance Interval," dated August 15, 2002 (ML022330608). The first was corrosion of the steel containment vessel due to degradation of the moisture barrier at the concrete floor to containment vessel interface. At the moisture barrier interface there have been small areas of surface corrosion and minor pitting detected; however, it does not represent an issue considering the available design margin. The condition has been remediated. The continued inspection of the moisture barrier is not considered an augmented inspection as it is adequately addressed by the inspection requirements established in Table IWE-2500-1 Item No. E1.30. The second condition involved external corrosion of the component cooling water containment penetrations due to condensation. Corrosion removal, recoating, and installation of insulation has been completed on the most affected penetrations. The area where these containment penetrations are located has been designated for augmented inspection as areas 2 and 4 (listed previously). These conditions were determined to have no significant degradation that could affect the containment function. Areas 1 and 3 (listed previously) were selected for augmented inspection based on the potential for moisture intrusion and the potential for coatings damage due to material handling in the vicinity of the equipment hatch.

3.5 Plant-Specific Confirmatory Analysis

3.5.1 Methodology

An evaluation has been performed to assess the risk impact of extending the St. Lucie ILRT interval from the current 10 years to 15 years. This plant-specific risk assessment follows the guidance of NEI 94-01, Revision 3-A, the methodology described in EPRI TR-1009325, Revision 2-A, and the NRC regulatory guidance outlined in RG 1.174 on the use of Probabilistic Risk Assessment (PRA) risk insights in support of a request to change plant licensing bases. In addition, the methodology used for Calvert Cliffs Nuclear Power Plant (considered a sister plant to St. Lucie by design) to estimate the likelihood and risk implication of corrosion-induced leakage of steel containment liners going undetected during the ILRT interval was also used in this analysis.

The St. Lucie Level 1 and simplified Level 2 internal events PRA model was used to perform the plant specific assessment. The analysis includes an evaluation for the dominant external events using information from the St. Lucie Individual Plant Examination of External Events (IPEEE). The St. Lucie IPEEE event models have not been updated since the original IPEEE. While a Fire PRA model has been developed to support transition to National Fire Protection Association (NFPA) NFPA 805, this Fire PRA model corresponds to the post-transition plant and does not match the current licensing basis for fire response. The NFPA 805 Fire PRA model has been undergoing a series of Requests for Additional Information (RAIs) and is still evolving within the review process. While the IPEEE models have not been updated, they include the pertinent information and insights and have, therefore, been used to estimate the effect on total LERF due to external events in the ILRT interval extension risk assessment.

By NRC letter dated June 25, 2008, the SE for EPRI Report No. 1009325 was transmitted and included the following terms from Section 4.2, "Limitations and Conditions," to be addressed.

Limitation/Condition Section 4.2 of SE for Revision 2	**FPL 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.	Refer to Section 3.5.2 of this submittal.
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 previous one-time 15-year ILRT extension requests. This would require the increase in CCFP be less than or equal to 1.5 percentage point. While acceptable for this application, the NRC staff is not endorsing these threshold values for other applications. Consistent with this limitation and condition, EPRI Report No. 1009325 will be revised in the "-A" version of the report, to change the population dose acceptance guidelines and the CCFP guidelines.	EPRI Report No. 1009325, Revision 2-A incorporates these population dose and Conditional Containment Failure Probability (CCFP) acceptance guidelines, and these guidelines have been used in the St. Lucie plant-specific risk impact assessment.
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 for the pre-existing containment large leak rate accident case (accident case 3b) used by the licensee 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 St. Lucie plant-specific risk impact assessment.
4. A LAR is required in instances where containment overpressure is relied upon for ECCS performance.	St. Lucie does not credit containment overpressure for net positive suction head of the ECCS pumps.

3.5.2 PRA Quality

The St. Lucie Internal Events PRA is composed of a Level 1 and Level 2 model for internal events (collectively called Internal Events PRA Model). The model was originally developed to support the Individual Plant Examination for Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities"; however, the model is maintained and upgraded in accordance with St. Lucie procedures. The models routinely incorporate review comments, current plant design, current procedures, plant operating data, current industry PRA techniques, and general improvements identified by the NRC.

The St. Lucie model has been updated and upgraded to meet Category II of the ASME/ANS PRA Standard as endorsed by Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The model was subjected to one full peer review followed by several focused peer reviews following each upgrade. The most recent self-assessment ensured compliance with RG 1.200, Revision 2 by identifying gaps in relation to the PRA standard as endorsed by the Regulatory Guide. When the identified gaps are resolved, LERF value is expected to be reduced. Thus, it is concluded that the current model is considered conservative with respect to LERF as applied to ILRT. As such, the St. Lucie PRA model is considered acceptable for use in assessing the risk impact of extending St. Lucie Units 1 and 2 reactor containment ILRT intervals to 15 years.

3.5.3 Summary of Plant-Specific Risk Assessment Results

Based on the results from the risk impact assessment and the associated sensitivity calculations, the following conclusions regarding the assessment of plant risk are associated with permanently extending the Type A ILRT test frequency to once in 15 years.

- RG 1.174 provides guidance for determining the risk impact of plant specific changes to the licensing basis. RG 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 core damage frequency (CDF), the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in 15 years is conservatively estimated as 4.49E-08/yr for Unit 1 and 6.28E-08/yr for Unit 2 using the EPRI guidance as written. As such, the estimated change in LERF for Unit 1 and Unit 2 is determined to be "very small" using the acceptance guidelines of RG 1.174.

- RG 1.174 also states that when the calculated increase in LERF is in the range of 1.00E-07 per reactor year to 1.00E-06 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.00E-05 per reactor year. An additional assessment of the impact from external events was also made. In this case, the total class 3b contribution to LERF including external events was conservatively estimated as 6.08E-07/yr for St. Lucie Unit 1 and 7.95E-07/yr for St. Lucie Unit 2. These are within the RG 1.174 acceptance criterion for total LERF of 1.00E-05/yr and, therefore, this change satisfies both the incremental and absolute expectations with regard to the RG 1.174 LERF metric.

- The change in Type A test frequency to once per 15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 2.67E-02 person-rem/yr for Unit 1 and 3.74E-02 person-rem/yr for Unit 2. Note that these values are based on internal events only and do not consider external events. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of $\leq$1.0 person- rem per year or $\leq$1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This is consistent with the NRC final SE for NEI 94-01 and EPRI Report No. 1009325. Moreover, the risk impact when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure probability from the three in 10 years interval to a permanent one time in 15 years interval is 0.83% for Unit 1 and 0.90% for Unit 2. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of $\leq$1.5 percentage points are very small. This is consistent with the NRC final SE for NEI 94-01 and EPRI Report No. 1009325. Accordingly, these increases are judged to be very small.

Based on these results, permanently increasing the ILRT interval to once in 15 years is considered a very small change to the St. Lucie risk profile. Details of the risk impact assessment are included as Attachment 5.

3.6 Conclusion

NEI 94-01, Revision 3-A describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A test (ILRT) intervals to 15 years. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. FPL is proposing to adopt the guidance of NEI 94-01, Revision 3-A, for the St. Lucie 10 CFR 50 Appendix J, Containment Leakage Rate Testing Program.

Based on the previous ILRTs conducted at St. Lucie, it is concluded that the extension of the containment ILRT interval from 10 to 15 years represents a minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with 10 CFR 50 Appendix J, Option B, and inspection activities under ASME Section XI IWE as performed under the St. Lucie ISI-IWE Program.

This experience is supplemented by risk analysis studies, including the St. Lucie risk impact assessment provided in Attachment 5. The findings of the St. Lucie risk impact assessment confirm the general finding 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 St. Lucie risk profile.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J of 10 CFR 50, "Primary Reactor Containment Testing for Water-Cooled Power Plants." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing, and reporting requirements for each type of test.

RG 1.163 was developed to endorse NEI 94-01, Revision 0 with certain modifications and additions.

The adoption of the Option B performance-based containment leakage rate testing for Type A testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, the test frequency is based upon an evaluation that reviews "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage.

NEI 94-01, Revision 3-A describes an approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. The document incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate test frequencies. In the SE issued by NRC letter dated June 8, 2012, the NRC concluded that NEI 94-01, Revision 3 describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE.

In an NRC letter dated August 20, 2013 (ML13192A394), the NRC requested that NEI 94-01, Revision 3 be updated as follows: "Revision 3 can be improved by stating that 'TR NEI 94-01, Revision 3, as modified by conditions and limitations in Section 4.0 of the NRC SE for Revision 2 and in Section 4.0 of the NRC SE for Revision 3, is acceptable for referencing as the implementing document for meeting the performance-based requirements of 10 CFR 50, Appendix J, Option B.'"

EPRI TR-1009325, Revision 2, provides a risk impact assessment for optimized ILRT intervals up to 15 years, utilizing current industry performance data and risk-informed guidance. NEI 94-01, Revision 3-A, states that a plant-specific risk impact assessment should be performed using the approach and methodology described in TR-1009325, Revision 2 for a proposed extension of the ILRT interval to 15 years. In the SE issued by NRC letter dated June 25, 2008, 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.

Based on the considerations 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 continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, FPL has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements or criteria.

4.2 No Significant Hazards Consideration

A change is proposed to St. Lucie Unit 1 and Unit 2 Technical Specification 6.8.4.h, "Containment Leakage Rate Testing Program." The proposed amendment would replace the reference to Regulatory Guide (RG) 1.163 with reference to Nuclear Energy Institute (NEI) Topical Report 94-01, Revision 3-A, dated July 2012, as the implementing document used by Unit 1 and Unit 2 to develop the St. Lucie performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. The proposed amendment would also extend the interval for the primary containment integrated leak rate test (ILRT) from 10 years to no longer than 15 years from the last ILRT.

Florida Power & Light Company (FPL) 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 change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change to the St. Lucie Unit 1 and Unit 2 Containment Leakage Rate Testing Program does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary reactor containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the primary containment itself and the testing requirements to periodically demonstrate the integrity of the primary containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve any accident precursors or initiators; therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

   The integrity of the reactor building is subject to two types of failure mechanisms which 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 itself combined with the periodic containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and regulatory commitments serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

   The proposed change adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for development of the St. Lucie performance-based testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less than the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably very small and determined to be within the guidelines published in RG 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. FPL has determined that the increase in Conditional Containment Failure Probability due to the proposed change would be very small; therefore, it is concluded that the proposed change does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that the proposed change does not involve 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 change adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A for the development of the St. Lucie performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. The primary containment and the testing requirements to periodically demonstrate the integrity of the primary containment exist to ensure the plant's ability to mitigate the consequences of an accident and 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.

   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 change adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A for the development of the St. Lucie performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. This change 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 Containment Leakage Rate Testing Program, as defined in the Technical Specifications (TS), ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests will be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A.

   Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is not detectable by an ILRT. A risk impact assessment using the current St. Lucie Unit 1 and Unit 2 risk models concluded that extending the ILRT test interval from 10 years to 15 years results in an acceptably small change to the St. Lucie risk profile.

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

4.3 Conclusion

Based on the above discussions, FPL concludes that the proposed amendment presents no 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.

5.0 ENVIRONMENTAL CONSIDERATIONS

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 change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Consequently, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENT

This request is similar to the license amendment requests submitted by other nuclear plants that were ultimately approved by the NRC:

- Virgil C. Summer Nuclear Station Unit 1, February 5, 2014 (ML13326A204)
- Arkansas Nuclear One Unit 2, April 7, 2011 (ML110800034)
- Palisades Nuclear Plant, April 23, 2012 (ML120740081)

7.0 REFERENCES

1. NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012 (ML12221A202)

Attachment 1

St. Lucie Unit 1 Technical Specifications Markup

(2) conform to the guidance of Appendix I to 10 CFR Part 50, and
(3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program 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 is in accordance with the guidelines contained ~~in Regulatory Guide 1.163, "Performance Based Containment Leak-Test Program," as modified by the following exception(s):~~ INSERT A

~~a) Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by R.G. 1.163) will be used for type A testing.~~

~~b) The first Type A test performed after the May 1993 Type A test shall be no later than May 2008.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 42.8 psig. The containment design pressure is 44 psig.

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests, $\leq 0.75\ L_a$ for Type A tests, and $\leq 0.096\ L_a$ for secondary containment bypass leakage paths.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For the personnel air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq 1.0\ P_a$.

3) For the emergency air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq$ 10 psig.

Unit 1 Technical Specifications Insert

Insert A

NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 8, 2005 Type A test shall be performed no later than December 8, 2020.

Attachment 2

St. Lucie Unit 2 Technical Specifications Markup

than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

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

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program 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 is in accordance with the guidelines contained in INSERT B ~~Regulatory Guide 1.163, "Performance Based Containment Leak Test Program," as modified by Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by R.G. 1.163) which will be used for type A testing.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 43.48 psig. The containment design pressure is 44 psig.

The maximum allowed ~~allow~~ containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Unit 2 Technical Specifications Insert

Insert B

NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 18, 2007 Type A test shall be performed no later than December 18, 2022.

Attachment 3

St. Lucie Unit 1 Retyped Technical Specifications Page

(2) conform to the guidance of Appendix I to 10 CFR Part 50, and
(3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program 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 is in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 8, 2005 Type A test shall be no later than December 8, 2020.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 42.8 psig. The containment design pressure is 44 psig.

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests, $\leq 0.75\ L_a$ for Type A tests, and $\leq 0.096\ L_a$ for secondary containment bypass leakage paths.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For the personnel air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq 1.0\ P_a$.

3) For the emergency air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to ≥ 10 psig.

Attachment 4

St. Lucie Unit 2 Retyped Technical Specifications Page

than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

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

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program 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 is in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 18, 2007 Type A test shall be performed no later than December 18, 2022.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 43.48 psig. The containment design pressure is 44 psig.

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Attachment 5

Permanent ILRT Interval Extension Risk Impact Analysis
Engineering Change 280892

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1 Purpose of Analysis

1.1 Purpose

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) to a permanent fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for St. Lucie Unit 1 and Unit 2. The risk assessment follows the guidelines from NEI 94-01(Reference 1), the methodology used in EPRI TR-104285 (Reference 2), the NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 (Reference 3), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 4), the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion induced leakage of steel liners going undetected during the extended test interval (Reference 5), and the methodology used in EPRI 1009325, Revision 2-A (Reference 20).

1.2 Background

Revisions to 10 CFR 50, 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 the limiting containment leakage rate of 1La[1].

The basis for the current fifteen year test interval is provided in Section 11.0 of NEI 94-01, Revision 3-A, and was established in 2008. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 (Reference 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."

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 the containment leak rate testing. In that analysis, it was determined that for a representative PWR

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[1] La (percent/24 hours) is the maximum allowable leakage rate at pressure Pa (calculated peak containment internal pressure related to the design basis accident) as specified in the technical specifications.

plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for St. Lucie Unit 1 and Unit 2.

The Guidance provided in Appendix H of EPRI Report No. 1009325, "*Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,*" (Reference 20) for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

The St. Lucie containment vessel is an independent free standing steel shell structure with a net free volume of approximately 2.5×10^6 ft^3. The St. Lucie Containment vessels are examined in accordance with the requirements of ASME Code Section XI, Subsection IWE, the plant protective coatings program, and Technical specifications.

The Containment Inservice Inspection program at PSL Units 1 and 2 is described in detail in 2nd Interval-IWE-PSL-1/2-Program-Plan, "Metal Containment Inservice Inspection Program" which provides the rules and requirements. The specific areas and components scheduled for inspection in accordance with the program are provided in 2nd Interval-IWE-PSL-1-Schedule, "ASME Section XI, Subsection IWE Containment Building Metal Containment Inservice Inspection Plan for St Lucie Unit 1" and 2nd Interval-IWE-PSL-2-Schedule, "ASME Section XI, Subsection IWE Containment Building Metal Containment Inservice Inspection Plan for St Lucie Unit 2." The program requirements include inspection of Containment Surfaces, Pressure Retaining Welds, Bolting, Seals, Gaskets, and Moisture Barriers using visual, surface, and volumetric techniques as required. Examinations that detect flaws or evidence of degradation shall be documented through the Condition Report process and dispositioned in accordance with the requirements IWE-3000. The IWE program performs inspection of the entire accessible interior surface of the containment in each of three periods within a ten year surveillance interval. One third of the moisture barrier at the concrete floor to vessel interface on both sides of containment is inspected during each period.

Inspection results indicate that no significant corrosion effects have been experienced on the containment vessel. At the moisture barrier interface there have been small areas of surface corrosion and minor pitting detected. However, it does not represent an issue considering the available design margin.

During activities that require repair of the containment vessel coatings, ASME Section XI, Subsection IWE requires visual exams to assess the condition of the vessel metal surface for evidence of flaking, blistering, peeling, discoloration and other signs of distress. Prior to any repair, an inspection is performed by NDE personnel to assess the condition of the base material. Following completion of coating repairs a final inspection is performed by NDE personnel to determine acceptability of the final condition and to act as a reference for future inspections.

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. The associated change to NEI 94-01 will require that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. 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. General visual inspections of both sides of the accessible containment vessel surface and the Shield Building are performed as required by Technical Specifications in accordance with Quality Instruction QI 10-PR/PSL-5, Technical Specification Surveillance Inspection of Reactor Building. Results of these inspections have not revealed any significant degradation of the containment shell.

1.3 Criteria

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this permanent extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in Core Damage Frequency (CDF) less than 10^{-6} per reactor year and increases in Large Early Release Frequency (LERF) less than 10^{-7} per reactor year. As St. Lucie Unit 1 and Unit 2 do not credit containment overpressure for the mitigation of design basis accidents, the Type A test does not impact CDF. Therefore, the relevant risk metric is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the Conditional Containment Failure Probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated. The criteria described below are taken from the NRC final safety evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 24).

Regarding CCFP, the NRC concluded that a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one time fifteen year ILRT extension requests. To this end the NRC has endorsed a small increase in CCFP as an increase in CCFP be less than or equal to 1.5% (Reference 24).

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. The NRC concluded that for purposes of assessing the risk impacts of the Type A ILRT extension in accordance with the EPRI methodology, 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 (Reference 24).

2 Methodology

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in Appendix H of EPRI Report No. 1009325, Revision 2-A, "*Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*" (Reference 20), EPRI TR-104285 (Reference 2), NUREG-1493 (Reference 6) and the Calvert Cliffs liner corrosion analysis (Reference 5). The analysis uses results from the current St. Lucie Unit 1 and Unit 2 Level 2 PRA models to establish frequency of fission product releases. Fission product release magnitudes are extrapolated from results of NUREG/CR-4551 to account for plant specific characteristics. This risk assessment is applicable to St. Lucie Unit 1 and Unit 2.

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report No. 1009325, Revision 2-A (Reference 20).

2. Develop plant specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.

3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.

4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 (Reference 4) and compare with the acceptance guidelines of RG 1.174.

5. Determine the impact of the ILRT interval extension on the Conditional Containment Failure Probability (CCFP) and the population dose and compare with the acceptance guidance of Reference 24.

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore:

- Consistent with the other industry containment leak risk assessments, the St. Lucie Unit 1 and Unit 2 assessment uses LERF and delta LERF in accordance with the risk acceptance guidance of RG 1.174. Changes in population dose and conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.

- The evaluation for St. Lucie Unit 1 and Unit 2 uses ground rules and methods to calculate changes in risk metrics that are similar to those used in Appendix H of EPRI Report No. 1009325, Revision 2-A, "*Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals.*"

3 Ground Rules

The following ground rules are used in the analysis:

- The technical adequacy of the St. Lucie 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.

- The current St. Lucie Unit 1 and Unit 2 Level 1 and Level 2 internal events PRA models are explicitly used in this analysis to assess fission product release frequencies.

- It is appropriate to use the St. Lucie Unit 1 and Unit 2 internal events PRA models as gauges to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations; this is evaluated in the sensitivity analysis which uses available information from the St. Lucie IPEEE (Reference 23) and EPU submittals (Reference 25 and Reference 26).

- Dose results for the containment failures modeled in the PRA can be characterized by scaling information provided in NUREG/CR-4551 (Reference 7). Specifically, St. Lucie population dose estimates are obtained by scaling the NUREG/CR-4551 reference plant results by differences in population, reactor power level (assumed proportional to fission product inventory), and nominal containment maximum leakage rate (La). Results of sensitivity studies are included which utilize St. Lucie Unit 1 and 2 release class doses used in the one-time 16 year ILRT extension (Reference 27), modified to account for differences in ILRT methodology and appropriately adjusted for power level and population growth (see discussion in Section 6.4)

- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology (Reference 2) and are summarized in Section 4.2.

- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.

- The representative containment leakage for Class 3a sequences is 10La based on the previously approved methodology performed for Indian Point Unit 3 (Reference 8 and Reference 9).

- The representative containment leakage for Class 3b sequences is 100La based on the guidance provided in EPRI Report No. 1009325, Revision 2-A.

- The Class 3b is very conservatively categorized as LERF based on the previously approved methodology (References 8 and 9).

- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

4 Inputs

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 (Reference 10)
2. NUREG/CR-4220 (Reference 11)
3. NUREG-1273 (Reference 12)
4. NUREG/CR-4330 (Reference 13)
5. EPRI TR-105189 (Reference 14)
6. NUREG-1493 (Reference 6)
7. EPRI TR-104285 (Reference 2)
8. NUREG-1150 (Reference 15) and NUREG/CR-4551 (Reference 7)
9. NEI Interim Guidance (Reference 3, Reference 17)
10. Calvert Cliffs Liner Corrosion Analysis (Reference 5)
11. EPRI Report No. 1009325, Revision 2-A, Appendix H (Reference 20)

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA 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 St. Lucie Unit 1 and Unit 2. 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 fifteen year extension of the ILRT interval.

4.1.1 NUREG/CR-3539 (Reference 10)

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400

(Reference 16) as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

4.1.2 NUREG/CR-4220 (Reference 11)

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.

4.1.3 NUREG-1273 (Reference 12)

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.

4.1.4 NUREG/CR-4330 (Reference 13)

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."

4.1.5 EPRI TR-105189 (Reference 14)

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.

4.1.6 NUREG-1493 (Reference 6)

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 three per ten years to one per twenty 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.

4.1.7 EPRI TR-104285 (Reference 2)

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 ..."

4.1.8 NUREG-1150 (Reference 15) and NUREG/CR 4551 (Reference 7)

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 St. Lucie Unit 1 and Unit 2 Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent St. Lucie. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

4.1.9 NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals (Reference 3, Reference 17)

The guidance provided in this document builds on the EPRI risk impact assessment methodology (Reference 2) and the NRC performance-based containment leakage test program (Reference 6), and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

4.1.10 Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension (Reference 5)

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, 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. Note that Calvert Cliffs is constructed with a steel lined concrete containment whereas the St. Lucie containments are free standing steel shells. The free standing steel shell design allows inspection of both sides of the steel plate above the basemat, thus reducing the potential for undetected corrosion.

4.1.11 EPRI Report No. 1009325, Revision 2-A, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (Reference 20)

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 (Reference 2) and the NRC performance-based containment leakage test program (Reference 6), and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the St. Lucie Unit 1 and Unit 2 assessments to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Classes 3a and 3b scenarios in this analysis as described in Section 5. The St. Lucie Unit 1 and 2 16 year ILRT extension used an early version of this methodology.

4.2 Plant Specific Inputs

The plant specific information used to perform the St. Lucie Unit 1 and Unit 2 ILRT Extension Risk Assessments include the following:

- Level 1 Model results
- Level 2 Model results
- Release category definitions used in the Level 2 Model
- Population within a 50 mile radius for the year 2040 which is based on an extrapolation of the 2000 census
- Containment Fragility Curves

4.2.1 Level 1 Model

The Level 1 PRA models that are used for St. Lucie Unit 1 and Unit 2 are characteristic of the as-built plants. The current Unit 1 Level 1 model is a linked fault tree model, and was quantified with the total Core Damage Frequency (CDF) = 5.34E-06/yr using a truncation value of 1.00E-13. The current Unit 2 Level 1 model is a linked fault tree model, and was quantified with the total Core Damage Frequency (CDF) = 6.77E-06/yr a truncation value of 1.00E-13.

4.2.2 Level 2 Model

The Level 2 Models that are used for St. Lucie Unit 1 and Unit 2 were developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 4-1 and Table 4-2 summarize the pertinent St. Lucie Unit 1 and Unit 2 results in terms of release category. Note that the enumerated total internal events Level 2 release frequency is slightly larger than that of the internal events CDF. This difference arises as a result of the numerical truncation issues resulting from the full integration of core damage end-states into the Level 2 model and the impact of the CAFTA small number approximation as applied to the detailed containment failure model. The small number approximation is a standard modeling practice. While this difference is observable, it does not significantly impact the results of the simplified Level 2 PRA or the associated conclusions drawn with regard to the ILRT extension.

Table 4-1: St. Lucie Unit 1 and Unit 2 Level 2 LERF Release Categories and Frequencies			
Release Category	**Definition**	**Unit 1 Frequency/yr[1]**	**Unit 2 Frequency/yr[1]**
LERF01	Non-SBO with a Low Pressure CFE	6.73E-09	1.51E-08
LERF02	Non-SBO with a High Pressure CFE	1.68E-11	1.75E-12
LERF03	Non-SBO with a Low Pressure CFE	5.94E-09	7.29E-10
LERF04	Non-SBO with a TI-SGTR	3.32E-09	4.13E-10
LERF05	Non-SBO with a Low Pressure CFE	1.45E-07	2.24E-08
LERF06	Non-SBO with a PI-SGTR	1.17E-08	6.08E-10
LERF07	Non-SBO with a Low Pressure CFE	3.02E-08	4.83E-08
LERF08	Non-SBO with Containment Isolation Failure	5.52E-08	7.47E-08
LERF09	Non-SBO with a Large Bypass Event	4.51E-07	3.25E-08
LERF10	SBO with a Low Pressure CFE	0.00E+00	0.00E+00
LERF11	SBO with a High Pressure CFE	7.38E-11	2.38E-11
LERF12	SBO with a Low Pressure CFE	2.02E-08	9.34E-09
LERF13	SBO with a TI-SGTR	3.89E-08	2.54E-08
LERF14	SBO with a Low Pressure CFE	6.17E-10	2.78E-10
LERF15	SBO with a PI-SGTR	7.76E-09	1.48E-09
LERF16	SBO with a Low Pressure CFE	4.12E-14	1.97E-13
LERF17	SBO with Containment Isolation Failure	2.90E-09	1.13E-09
LERF18	SBO with a Large Bypass Event	4.92E-13	5.20E-12
	Total LERF Release Category Frequency (LERF01 through LERF18)	7.79E-07	2.32E-07

Notes:

1. These values were quantified using a truncation value of 1.00E-14.

Table 4-2 summarizes all of the Level 2 release categories and frequencies. The CDF including uncategorized releases is determined by adding together all Level 2 release categories.

Table 4-2: St. Lucie Unit 1 and Unit 2 Level 2 Release Categories and Frequencies			
Release Category	Definition	Unit 1 Frequency/yr[1]	Unit 2 Frequency/yr[1]
INTACT	Containment Intact	4.22E-06	5.84E-06
SERF	Small Early Release	7.07E-08	1.56E-07
LATE	Late Release	3.69E-07	7.04E-07
LERF	Total Large Early Releases	7.79E-07	2.32E-07
CDF (including uncategorized releases)		5.44E-06[2]	6.93E-06[2]

Notes:
1. These values were quantified using a truncation value of 1.00E-14.
2. This value was calculated as a sum of all release categories (INTACT, SERF, LATE, and LERF).

4.2.3 Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results to reflect the demographics around St. Lucie Unit 1 and Unit 2. Each of the release categories from Table 4-1 was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551 (see below). The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 7 bins that are relevant to the analysis. The definitions of the 7 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 4-3 for reference purposes. Table 4-4 summarizes the calculated population dose for Surry associated with each APB from NUREG/CR-4551.

Table 4-3: Summary Accident Progression Bin (APB) Descriptions (Reference 7)	
Summary APB Number	Description
1	CD, VB, Early CF, Alpha Mode Core damage occurs followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment except those follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	CD, VB, Early CF, RCS Pressure > 200 psia Core Damage occurs followed by vessel breach. Implies Early CF with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.

Table 4-3: Summary Accident Progression Bin (APB) Descriptions (Reference 7)

Summary APB Number	Description
3	CD, VB, Early CF, RCS Pressure < 200 psia Core damage occurs followed by vessel breach. Implies Early CF with the RCS below psia when the containment fails. It does not include bins in which the containment failure at VB or an SGTR, or Alpha mode failures.
4	CD, VB, Late CF Core Damage occurs followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	CD, Bypass Core Damage occurs followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.
6	CD, VB, No CF Core Damage occurs followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	CD, No VB, No CF Core Damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures except those that bypass the containment. Most of the bins placed in this reduce bin have no containment failure as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and the core is brought to a safe stable state before the vessel fails.

For the baseline analysis dose estimates are based on extrapolation of the results of the Surry assessment (Reference 7). For the purpose of sensitivity studies, this analysis uses St. Lucie doses that were established for the one-time 16 year ILRT extension and are adjusted to account for changes in power level, containment leakage and expected demographics. Population estimates are based on an extrapolation of the 2000 census data to year 2040.

Table 4-4: Calculation of Surry Population Dose Risk at 50 Miles (Reference 7)				
Collapsed Bin #	**Fractional APB Contributions to Risk (MFCR) (1)**	**NUREG/CR-4551 Population Dose Risk at 50 miles (person-rem/yr, mean) (2)**	**NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)**	**NUREG/CR-4551 Population Dose at 50 miles (person-rem) (4)**
1	0.029	0.158	1.23E-07	1.28E+06
2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.012E-08	6.46E+05 (5)
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	1.000	5.55	4.1E-05	

(1) Mean Fractional Contribution to Risk calculated from the average of two samples delineated in Table 5.1-3 of NUREG/CR-4551.
(2) The total population dose risk at 50 miles from internal events in person-rem is provided as the average of two samples in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.
(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-3. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.
(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.
(5) Assumed population dose at 50 miles for Collapsed Bin #3 equal to that of Collapsed Bin #2. Collapsed Bin Frequency #3 was then back calculated using that value. This does not influence the results of this evaluation since Bin #3 does not appear as part of the results for St. Lucie Unit 1 and Unit 2.

4.2.4 Population Dose Estimate Methodology

In accordance with Reference 1, the person-rem results in Table 4-4 can be used as an approximation of the dose for St. Lucie Unit 1 and Unit 2 if it is corrected for allowable containment leak rate (La), reactor power level and the population density surrounding St. Lucie.

<u>La adjustment:</u>

$$\text{FLeakage} = \frac{\text{La of St. Lucie Unit 1 and Unit 2 (\%w/o/day)}}{\text{La of reference plant (applicable only to those APBs affected by normal leakage)}}$$

La for St. Lucie Unit 1 and Unit 2 is 0.5%w/o/day (References 25 and 26). La for Surry is 0.1%w/o/day.

FLeakage = 0.5 / 0.1

FLeakage = 5

Power level adjustment:

$$\text{FPower} = \frac{\text{Rated power level of St. Lucie Unit 1 and Unit 2 (MWt)}}{\text{Rated power level of reference plant (MWt)}}$$

The rated power level for St. Lucie Unit 1 and Unit 2 is 3020 MWt (References 25 and 26). The rated power level for Surry is 2441MWt.

FPower = 3020 MWt / 2441 MWt

FPower = 1.237

Population density adjustment:

The total population within a 50 mile radius of St. Lucie Unit 1 and Unit 2 is 1.757E+06. This number is based on the extrapolation of the 2000 census including the impact of population growth projected for the year 2040. This population value is compared to the population value that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for St. Lucie Unit 1 and Unit 2.

Total 2040 estimated St. Lucie Unit 1 and Unit 2 Population within 50 miles = 1.757E+06

Surry Population within a 50 mile radius from the NUREG/CR-4551 reference plant = 1.23E+06

FPopulation = 1.757E+06 / 1.23E+06 = 1.428

The factors developed above are used to adjust the population dose for the surrogate plant (Surry) for St. Lucie Unit 1 and Unit 2. For intact containment endstates, the total population dose factor is as follows:

FIntact = FPopulation * FPower * FLeakage

FIntact = 1.428 * 1.237 * 5

FIntact = 8.836

For EPRI accident classes not dependent on containment leakage, the population dose factor is as follows:

FOthers = FPopulation * FPower

FOthers = 1.428 * 1.237

FOthers = 1.767

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. The above adjustments provide an approximation for St. Lucie Unit 1 and Unit 2 of the population doses associated with each of the release categories from NUREG/CR-4551.

Table 4-5 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for St. Lucie Unit 1 and Unit 2.

Table 4-5: Calculation of St. Lucie Unit 1 and Unit 2 Population Dose Risk at 50 Miles

Accident Progression Bin (APB)	NUREG/CR-4551 Population Dose at 50 miles (person-rem)	Bin Multiplier used to obtain St. Lucie Population Dose		St. Lucie Adjusted Population Dose at 50 miles (person-rem)
1	1.28E+06	FOther	1.767	2.26E+06
2	6.46E+05	FOther	1.767	1.14E+06
3	6.46E+05	FOther	1.767	1.14E+06
4	4.95E+05	FOther	1.767	8.75E+05
5	8.12E+05	FOther	1.767	1.43E+06
6	4.23E+02	FIntact	8.836	3.74E+03
7	5.76E+02	FIntact	8.836	5.09E+03

4.2.5 Application of St. Lucie Unit 1 and Unit 2 PRA Model Results to NUREG/CR-4551 Level 3 Output

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the St. Lucie Unit 1 and Unit 2 PRA Level 2 models are not defined in the same terms as reported in NUREG/CR-4551. The St. Lucie PRA Level 2 model results are defined as four main release categories including INTACT, SERF, LATE, and LERF. In order to use the Level 3 model presented in that document, it was necessary to match the St. Lucie PRA Level 2 release categories to the collapsed APBs. The St. Lucie Level 2 release categories and frequencies are from the current simplified Level 2 model. The assignments are shown in Table 4-6, along with the corresponding EPRI classes (see below). The EPRI classes and descriptions are listed in Table 4-7 in addition to the St. Lucie Level 2 release categories.

Table 4-6: St. Lucie Unit 1 and Unit 2 Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI Accident Classes

St. Lucie Level 2 Release Category Frequency	Unit 1 Frequency (per yr)	Unit 2 Frequency (per yr)	Definition	NUREG/CR -4551 APB	EPRI Class
INTACT	4.22E-06	5.84E-06	Containment Intact	6,7	1
SERF	7.07E-08	1.56E-07	Small Early Release	3	3
LATE	3.69E-07	7.04E-07	Late Release	4	7
LERF01	6.73E-09	1.51E-08	Non-SBO with a Low Pressure CFE	3	7
LERF02	1.68E-11	1.75E-12	Non-SBO with a High Pressure CFE	2	7
LERF03	5.94E-09	7.29E-10	Non-SBO with a Low Pressure CFE	3	7
LERF04	3.32E-09	4.13E-10	Non-SBO with a TI-SGTR	5	8
LERF05	1.45E-07	2.24E-08	Non-SBO with a Low Pressure CFE	3	7
LERF06	1.17E-08	6.08E-10	Non-SBO with a PI-SGTR	5	8
LERF07	3.02E-08	4.83E-08	Non-SBO with a Low Pressure CFE	3	7
LERF08	5.52E-08	7.47E-08	Non-SBO with Containment Isolation Failure	1	2
LERF09	4.51E-07	3.25E-08	Non-SBO with a Large Bypass Event	5	8
LERF10	0.00E+00	0.00E+00	SBO with a Low Pressure CFE	3	7
LERF11	7.38E-11	2.38E-11	SBO with a High Pressure CFE	2	7
LERF12	2.02E-08	9.34E-09	SBO with a Low Pressure CFE	3	7
LERF13	3.89E-08	2.54E-08	SBO with a TI-SGTR	5	8
LERF14	6.17E-10	2.78E-10	SBO with a Low Pressure CFE	3	7
LERF15	7.76E-09	1.48E-09	SBO with a PI-SGTR	5	8
LERF16	4.12E-14	1.97E-13	SBO with a Low Pressure CFE	3	7
LERF17	2.90E-09	1.13E-09	SBO with Containment Isolation Failure	1	2
LERF18	4.92E-13	5.20E-12	SBO with a Large Bypass Event	5	8

4.2.6 Release Category Definitions

Table 4-7 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology (Reference 2). These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

Table 4-7: EPRI Containment Failure Classification (Reference 2)		
Class	**Description**	**St. Lucie Level 2 Release Category Frequency**
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values La, under Appendix J for that plant	INTACT
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.	LERF08, LERF17
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.	SERF
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.	N/A
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.	N/A
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.	N/A
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.	LATE, LERF01, LERF02, LERF03, LERF05, LERF07, LERF10, LERF11, LERF12, LERF14, LERF16
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.	LERF04, LERF06, LERF09, LERF13, LERF15, LERF18

4.3 Impact of Extension on Detection of Component Failures That Lead to Leakage (Small and Large)

The ILRT can detect a number of component failures such as liner breach, failure of certain bellow arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 4-7, it is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the EPRI Guidance (Reference 20). For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 "small" failures in 217 tests leads to 2/217=0.0092). For Class 3b, Jefferys non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5 / (217+1) = 0.0023).

In a follow on letter (Reference 17) to their ILRT guidance document (Reference 3), NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC Regulatory Guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> *The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. 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 a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.*

The application of this additional guidance to the analysis for St. Lucie Unit 1 and Unit 2, as detailed in Section 5, involves the following:

- The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large preexisting containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the St. Lucie Unit 1 and Unit 2 Level 2 PRA analyses.

- Class 1 accident sequences may involve availability and or successful operation of containment sprays. It could be assumed that, for calculation of the Class 3b and 3a frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. However, in this assessment St. Lucie Unit 1 and Unit 2 do not credit containment spray as a means of reducing releases from Class 3 events.

Consistent with the NEI Guidance (Reference 3), the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten year interval is five years (10 yr/2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. An extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak compared to a three year interval.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC (Reference 9)) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

4.4 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis (Reference 5). The Calvert Cliffs analysis was performed for a concrete cylinder, dome and a concrete basemat, each with a steel liner. St. Lucie Unit 1 and Unit 2 are steel shell containments and are capable of visual inspection on both sides of the containment shell for approximately 80% of the containment surface (Reference 28).

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of a containment steel liner. It should be noted that this computation is being applied to provide an upper bound approach to quantify corrosion induced risk. Of the two industry events identified as a basis for the Calvert Cliffs corrosion risk assessment, neither is considered applicable to the St. Lucie containment designs. Furthermore, the likelihood of detection of significant corrosion for the 80% of the containment is very high. Regardless, the Calvert Cliffs corrosion likelihood methodology is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the upper containment (cylinder and dome regions in Calvert Cliffs evaluation)
- 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

4.4.1 Assumptions

- Consistent with the Calvert Cliffs analysis, a half failure is conservatively assumed for basemat concealed liner corrosion due to the lack of identified failures (See Table 4-8, Step 1).
- There are two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis. These events have been determined by NextEra to not be applicable to the St. Lucie Unit 1 and Unit 2 but are applied to the containment analyses. This is a conservative application as either side of the steel shell in the upper portion of the containment can be visually inspected and only 20% of containment is below the basement. The events included in the Calvert Cliffs corrosion assessment process, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the nonvisible (backside) portion of the containment liner.
- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is based on 70 steel-lined containments.
- The Calvert Cliffs analysis used the estimated historical liner flaw probability of 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date. Since the time of the Calvert Cliffs submittal, two additional relevant liner corrosion events involving concealed corrosion (corrosion initiated on the inaccessible liner surface) were observed and are considered in the corrosion risk assessment. These events occurred at Beaver Valley Unit 1 and D.C. Cook Unit 2 (Reference 21 and Reference 22, respectively). Consistent with the addition of the two observed events, the historical liner flaw probability was established by incrementing the flaw observation time by 7.75 years. This re-evaluation resulted in a reduction of the historical liner flaw likelihood to 4.3E-03/year ((2+2) / [70 * (5.5 + 7.75)] = 4.3E-03/year). This value is smaller than the value of 5.2E-03 which is used in the Calvert Cliffs analysis. The conservative value of 5.2E-03 will be used in this St. Lucie report to remain consistent with the Calvert Cliffs analysis.
- Consistent with the Calvert Cliffs analysis, the steel plate flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel ages. (See Table 4-8, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, 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. These values were determined from an assessment of the probability versus containment pressure, and the

selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For St. Lucie Unit 1 and Unit 2, the containment failure probabilities are less than these values at 37 psig based on the containment fragility curve which is documented in the St. Lucie Unit 1 and Unit 2 Level 2 analyses. A containment bypass model is utilized for LERF. Conservative probabilities of 1% for the shell above the basemat and 0.1% for the basemat are used in this analysis, and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude (See Table 4-8, Step 4).

- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the upper containment region (See Table 4-8, Step 4).

- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection (See Table 4-8, Step 5). Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.

- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

4.4.2 Analysis

<table>
<tr><th colspan="6">Table 4-8: Steel Liner Corrosion Base Case</th></tr>
<tr><th>Step</th><th>Description</th><th colspan="2">Upper Containment</th><th colspan="2">Containment Basemat</th></tr>
<tr><td>1</td><td>Historical Steel Liner Flaw Likelihood

Failure Data: Containment location specific</td><td colspan="2">Events: 2

(2)/(70 * 5.5) = 5.2E-03</td><td colspan="2">Events: 0
(assume half a failure)

0.5/(70 * 5.5) = 1.3E-03</td></tr>
<tr><td rowspan="5">2</td><td rowspan="5">Age Adjusted Steel Liner Flaw Likelihood

During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).</td><td>Year</td><td>Failure Rate</td><td>Year</td><td>Failure Rate</td></tr>
<tr><td>1</td><td>2.1E-03</td><td>1</td><td>5.0E-04</td></tr>
<tr><td>avg 5-10</td><td>5.2E-03</td><td>avg 5-10</td><td>1.3E-03</td></tr>
<tr><td>15</td><td>1.4E-02</td><td>15</td><td>3.5E-03</td></tr>
<tr><td colspan="2">15 year average =
6.27E-03</td><td colspan="2">15 year average =
1.57E-03</td></tr>
<tr><td>3</td><td>Flaw Likelihood at 3, 10, and 15 years

Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference 5).</td><td colspan="2">0.71% (1 to 3 years)
4.06% (1 to 10 years)
9.40% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the intervals of concern in this analysis.)</td><td colspan="2">0.18% (1 to 3 years)
1.02% (1 to 10 years)
2.35% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the intervals of concern in this analysis.)</td></tr>
</table>

Table 4-8: Steel Liner Corrosion Base Case			
Step	**Description**	**Upper Containment**	**Containment Basemat**
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 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 events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the leakages for the upper containment and the containment basemat as summarized below for St. Lucie Unit 1 and Unit 2.

Total Likelihood of Non-Detected Containment Leakage Due To Corrosion for St. Lucie Unit 1 and Unit 2:

At 3 years: 0.00071% + 0.00018% = 0.00089%

At 10 years: 0.0041% + 0.0010% = 0.0051%

At 15 years: 0.0094% + 0.0024% = 0.012%

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. For example, the three in ten year case is calculated as follows:

- Per Table 4-6, the St. Lucie Unit 1 and Unit 2 CDF associated with accidents that are not independently LERF or could never result in LERF are Level 2 Release Categories INTACT, SERF and LATE. Therefore the St. Lucie Unit 1 CDF associated with accidents that are not independently LERF or could never result in LERF is equal to 4.22E-06/yr + 7.07E-08/yr + 3.69E-07/yr = 4.66E-06/yr. The St. Lucie Unit 2 CDF associated with accidents that are not independently LERF or could never result in LERF is equal to 5.84E-06/yr + 1.56E-07/yr + 7.04E-07/yr = 6.70E-06/yr.

- Per Table 5-3, the EPRI Class 3b frequency is 1.12E-08/yr for Unit 1 and 1.56E-08/yr for Unit 2.

- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 4.66E-06/yr * 0.00089% = 4.15E-11/yr for Unit 1 and 6.70E-06/yr * 0.00089% = 5.97E-11/yr, where 0.00089% was previously shown above to be the cumulative likelihood of non-detected containment leakage due to corrosion at three years.

- The three in ten year Class 3b frequency including the corrosion-induced concealed flaw issue is then calculated as 1.12E-08/yr + 4.15E-11/yr = 1.12E-08/yr for Unit 1 and 1.56E-08/yr + 5.97E-11/yr = 1.57E-08/yr for Unit 2.

5 Results

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H, EPRI-TR-104285 (Reference 2) and previous risk assessment submittals on this subject (References 5, 8, 18, 19) have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5-1 lists these accident classes.

The analysis performed examined St. Lucie Unit 1 and Unit 2 specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown of the severe accidents contributing to risk were considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).

- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.

- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.

- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-1: Accident Classes

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (Liner Breach)
3b	Large Isolation Failures (Liner Breach)
4	Small Isolation Failures (Failure to Seal-Type B)
5	Small Isolation Failures (Failure to Seal-Type C)
6	Other Isolation Failures (e.g., Dependent Failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including Very Low and No Release)

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5-1.

Step 2 - Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact of extending Type A test interval from three to fifteen and ten to fifteen years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP).

5.1 Step 1 - Quantify the Base-Line Risk in Terms of Frequency Per Reactor Year

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5-1 were developed for St. Lucie Unit 1 and Unit 2 by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 4-6, scaling these frequencies to account for the uncategorized sequences, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion per the methodology described in Section 4.4.

For Unit 1, the total frequency of the categorized sequences is 5.37E-06/yr, the total CDF[2] is 5.44E-06/yr, and the scale factor is 1.013. The scaling factor is determined by dividing the total core damage frequency (including the uncategorized frequency) by the total categorized release category frequency (5.44E-06/yr / 5.37E-06/yr = 1.013). For Unit 2, the total frequency of the categorized sequences is 6.78E-06/yr, the total CDF is 6.93E-06/yr, and the scale factor is 1.023. The scaling factor is determined by dividing the total core damage frequency (including the uncategorized frequency) by the total categorized release category frequency (6.93E-06/yr / 6.78E-06/yr = 1.023).

This process ensures that the bounding CDF of 5.44E-06/yr for Unit 1 and 6.93E-06/yr for Unit 2 is maintained for the determination of Class 3 states (see below) and effectively distributes the dose impact of the non-represented classes (SERF endstates which are considered Classes 4, 5, and 6) proportionately (per frequencies identified in the adjusted columns of Table 5-2) over the evaluated Classes 1, 2, 7, and 8. The CDF values from Table 4-2 include all release categories (uncategorized results include Classes 4, 5, and 6). Table 5-2 contains the frequencies from the specific categorized sequences and the resulting frequencies due to the application of the scale factor (which redistributes the frequencies to the categorized endstates).

[2] CDF as established from the summation of the CAFTA Level 2 release classes (see Table 4-2).

Table 5-2: St. Lucie Unit 1 and Unit 2 Categorized Accident Classes and Frequencies

EPRI Class	St. Lucie Release Category	Unit 1		Unit 2	
		Frequency Based on Categorized Results (per yr)	Adjusted Frequency Using Scale Factor of 1.013 (per yr)	Frequency Based on Categorized Results (per yr)	Adjusted Frequency Using Scale Factor of 1.023 (per yr)
1	Intact Containment (INTACT)	4.22E-06	4.27E-06	5.84E-06	5.98E-06
2	Containment Isolation Failures (LERF08 & LERF17)	5.81E-08	5.89E-08	7.59E-08	7.76E-08
7	Late Containment Failure (LATE) and Containment Failure (LERF01, LERF02, LERF03, LERF05, LERF07, LERF10, LERF11, LERF12, LERF14, LERF16)	5.78E-07	5.86E-07	8.00E-07	8.19E-07
8	Containment Bypass (LERF09 & LERF18) and SGTR (LERF04, LERF06, LERF13 & LERF15)	5.12E-07	5.19E-07	6.04E-08	6.18E-08
Total Frequency		5.37E-06	5.44E-06	6.78E-06	6.93E-06

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially determined from the Containment Intact Level 2 Release Category listed in Table 4-6 minus the EPRI Class 3a and 3b frequency, which are calculated below.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Large Containment Isolation Failures Level 2 Release Category listed in Table 4-6.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (in excess of design allowable but <10La) or large (>100La).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage

= 0.0092 [see Section 4.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage

= 0.0023 [see Section 4.3]

As described in Section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions).

Class 3a Frequency (Unit 1) = 0.0092 * (CDF - (Class 2 + Class 8))

= 0.0092 * (5.44E-06/yr – (5.89E-08/yr + 5.19E-07/yr)) = 4.47E-08/yr

Class 3b Frequency (Unit 1) = 0.0023 * (CDF - (Class 2 + Class 8))

=0.0023 * (5.44E-06/yr – (5.89E-08/yr + 5.19E-07/yr)) = 1.12E-08/yr

Class 3a Frequency (Unit 2) = 0.0092 * (CDF - (Class 2 + Class 8))

= 0.0092 * (6.93E-06/yr – (7.76E-08/yr + 6.18E-08/yr)) = 6.25E-08/yr

Class 3b Frequency (Unit 2) = 0.0023 * (CDF - (Class 2 + Class 8))

=0.0023 * (6.93E-06/yr – (7.76E-08/yr + 6.18E-08/yr)) = 1.56E-08/yr

For this analysis, the associated containment leakage for Class 3a is 10La and for Class 3b is 100La. These assignments are consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

Note, in the above equations for the Class 3a and 3b release frequencies, the total adjusted release frequency from the appropriate columns of Table 5-2 has been substituted for CDF. As discussed previously this process marginally over-estimates the Class 3 releases.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type C test components occurs. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. 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. Consistent with guidance provided in EPRI Report No. 1009325, Revision 2-A, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from the Severe Accident Phenomena-Induced Failures Release Category from the St. Lucie Unit 1 and Unit 2 Level 2 results shown in Table 4-6.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from the Containment Bypass Release Category from the St. Lucie Unit 1 and Unit 2 Level 2 results shown in Table 4-6.

5.1.1 Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI-TR-104285 the NEI Interim Guidance, and guidance provided in EPRI Report No. 1009325, Revision 2-A. Table 5-3 summarizes these accident frequencies by accident class for St. Lucie Unit 1 and Unit 2.

Table 5-3: Radionuclide Release Frequencies as a Function of Accident Class (St. Lucie Unit 1 and Unit 2 Base Case)					
Accident Classes (Containment Release Type)	**Description**	**Unit1 Frequency (per Rx-yr)**		**Unit 2 Frequency (per Rx-yr)**	
		Base Case	**Base Case Plus Corrosion[1]**	**Base Case**	**Base Case Plus Corrosion[1]**
1	No Containment Failure	4.22E-06	4.22E-06	5.90E-06	5.90E-06
2	Large Isolation Failures (Failure to Close)	5.89E-08	5.89E-08	7.76E-08	7.76E-08
3a	Small Isolation Failures (liner breach)	4.47E-08	4.47E-08	6.25E-08	6.25E-08
3b	Large Isolation Failures (liner breach)	1.12E-08	1.12E-08	1.56E-08	1.57E-08
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	5.86E-07	5.86E-07	8.19E-07	8.19E-07
8	Bypass (Interfacing System LOCA)	5.19E-07	5.19E-07	6.18E-08	6.18E-08
CDF	All CET end states	5.44E-06	5.44E-06	6.93E-06	6.93E-06

1. Note that this is based on data developed in Section 4.4. Only Class 3b is impacted by the corrosion.

5.2 Step 2 - Develop Plant Specific Person-Rem Dose (Population Dose) Per Reactor Year

Plant specific release analyses were performed to estimate the person-rem doses to the population within a 50 mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 4.2, and summarized in Table 4-5. The results of applying these releases to the EPRI containment failure classification are as follows:

Class 1 = (3.74E+03 person-rem (at 1.0La) + 5.09E+03 person-rem (at 1.0La)) / 2 = 4.41E+03 person-rem (1)

Class 2 = 1.14E+06 person-rem (2)

Class 3a = 4.41E+03 person-rem x 10La = 4.41E+04 person-rem (3)

Class 3b = 4.41E+03 person-rem x 100La = 4.41E+05 person-rem (3)

Class 4 = Not analyzed

Class 5 = Not analyzed

Class 6 = Not analyzed

Class 7 = 8.75E+05 person-rem (4)

Class 8 = 1.43E+06 person-rem (5)

(1) The derivation is described in Section 4.2 for St. Lucie Unit 1 and Unit 2. Class 1 is assigned the dose from the "no containment failure" APBs from NUREG/CR-4551 (i.e., APB #6 and APB #7). The dose is calculated as an arithmetic average of the dose for these bins and is bounding[3].

(2) The Class 2, containment isolation failures, dose is assigned from APB #2 (Early CF).

(3) The Class 3a and 3b dose are related to the Class 1 leakage rate as shown. While no pre-existing leakage in excess of 21 La has been identified for any historical ILRT event, Class 3b releases are conservatively assessed at 100La. Class 3a releases are conservatively assessed at 10La. This is consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

(4) The Class 7 dose is assigned from APB #4 (Late CF)[4].

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from APB #5 (Bypass).

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology (Reference 2) containment failure classifications, and consistent with the NEI guidance (Reference 3) as modified by EPRI Report No. 1009325, Revision 2-A are provided in Table 5-4.

[3] The use of a simple average dose is bounding as the average over-estimates the proportion of endstates that initiated with a pre-existing isolation failure (APB #7) in the St. Lucie Level 2 model. These states would be classified as SERFs.

[4] States 3 and 4 map into EPRI release class 7, however, state 3 represents 1% of the contributors to this release class. The dose is selected based on the LATE APB only is conservative as the value does not directly impact the Class 3 doses and is used as an element in the baseline dose for use in fractional dose comparisons.

Table 5-4: St. Lucie Unit 1 and Unit 2 Population Dose Estimates for Population Within 50 Miles

Accident Classes (Containment Release Type)	Description	Unit 1 and Unit 2 Person-Rem (50 miles)
1	No Containment Failure	4.41E+03
2	Large Isolation Failures (Failure to Close)	1.14E+06
3a	Small Isolation Failures (liner breach)	4.41E+04
3b	Large Isolation Failures (liner breach)	4.41E+05
4	Small Isolation Failures (Failure to seal-Type B)	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	8.75E+05
8	Bypass (Interfacing System LOCA)	1.43E+06

The above dose estimates, when combined with the results presented in Table 5-3, yield the St. Lucie Unit 1 and Unit 2 baseline mean consequence measures for each accident class. These results are presented in Table 5-5 and Table 5-6.

Table 5-5: St. Lucie Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person- Rem/yr (50 miles)	
1	No Containment Failure (2)	4.41E+03	4.22E-06	1.86E-02	4.22E-06	1.86E-02	-1.83E-07
2	Large Isolation Failures (Failure to Close)	1.14E+06	5.89E-08	6.72E-02	5.89E-08	6.72E-02	0.00E+00
3a	Small Isolation Failures (liner breach)	4.41E+04	4.47E-08	1.97E-03	4.47E-08	1.97E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	4.41E+05	1.12E-08	4.93E-03	1.12E-08	4.95E-03	1.83E-05
4	Small Isolation Failures (Failure to seal -Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	8.75E+05	5.86E-07	5.12E-01	5.86E-07	5.12E-01	0.00E+00

Table 5-5: St. Lucie Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person- Rem/yr (50 miles)	
8	Bypass (Interfacing System LOCA)	1.43E+06	5.19E-07	7.45E-01	5.19E-07	7.45E-01	0.00E+00
CDF	All CET end states	N/A	5.44E-06	1.35E+00	5.44E-06	1.35E+00	1.81E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-6: St. Lucie Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person- Rem/yr (50 miles)	
1	No Containment Failure (2)	4.41E+03	5.90E-06	2.60E-02	5.90E-06	2.60E-02	-2.63E-07
2	Large Isolation Failures (Failure to Close)	1.14E+06	7.76E-08	8.86E-02	7.76E-08	8.86E-02	0.00E+00
3a	Small Isolation Failures (liner breach)	4.41E+04	6.25E-08	2.76E-03	6.25E-08	2.76E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	4.41E+05	1.56E-08	6.90E-03	1.57E-08	6.92E-03	2.63E-05
4	Small Isolation Failures (Failure to seal -Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	8.75E+05	8.19E-07	7.16E-01	8.19E-07	7.16E-01	0.00E+00

Table 5-6: St. Lucie Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person- Rem/yr (50 miles)	
8	Bypass (Interfacing System LOCA)	1.43E+06	6.18E-08	8.87E-02	6.18E-08	8.87E-02	0.00E+00
CDF	All CET end states	N/A	6.93E-06	9.29E-01	6.93E-06	9.29E-01	2.61E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.3 Step 3 - Evaluate Risk Impact of Extending Type A Test Interval From 10 to 15 Years

The next step is to evaluate the risk impact of extending the test interval from its current ten year value to fifteen years. To do this, an evaluation must first be made of the risk associated with the ten year interval since the base case applies to a three year interval (i.e., a simplified representation of a three in ten interval).

5.3.1 Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is directly impacted. As it is assumed that the new Class 3 endstates arise from previously intact containment states, the intact state frequency is reduced accordingly. The risk contribution is changed based on the NEI guidance as described in Section 4.3 by a factor of 3.33 compared to the base case values. The Unit 1 and Unit 2 results of the calculation for a ten year interval are presented in Table 5-7 and Table 5-8, respectively.

5.3.2 Risk Impact Due to 15-Year Test Interval

The risk contribution for a fifteen year interval is calculated in a manner similar to the ten year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the three year interval value, as described in Section 4.3. The Unit 1 and Unit 2 results for this calculation are presented in Table 5-9 and Table 5-10, respectively.

Table 5-7: St. Lucie Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person-Rem/yr (50 miles)**	
1	No Containment Failure (2)	4.41E+03	4.09E-06	1.80E-02	4.09E-06	1.80E-02	-6.09E-07
2	Large Isolation Failures (Failure to Close)	1.14E+06	5.89E-08	6.72E-02	5.89E-08	6.72E-02	0.00E+00
3a	Small Isolation Failures (liner breach)	4.41E+04	1.49E-07	6.57E-03	1.49E-07	6.57E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	4.41E+05	3.72E-08	1.64E-02	3.74E-08	1.65E-02	6.09E-05
4	Small Isolation Failures(Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	8.75E+05	5.86E-07	5.12E-01	5.86E-07	5.12E-01	0.00E+00

Table 5-7: St. Lucie Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
8	Bypass (Interfacing System LOCA)	1.43E+06	5.19E-07	7.45E-01	5.19E-07	7.45E-01	0.00E+00
CDF	All CET end states	N/A	5.44E-06	1.37E+00	5.44E-06	1.37E+00	6.03E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-8: St. Lucie Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	4.41E+03	5.72E-06	2.52E-02	5.72E-06	2.52E-02	-8.77E-07
2	Large Isolation Failures (Failure to Close)	1.14E+06	7.76E-08	8.86E-02	7.76E-08	8.86E-02	0.00E+00
3a	Small Isolation Failures (liner breach)	4.41E+04	2.08E-07	9.19E-03	2.08E-07	9.19E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	4.41E+05	5.20E-08	2.30E-02	5.22E-08	2.31E-02	8.77E-05
4	Small Isolation Failures(Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	8.75E+05	8.19E-07	7.16E-01	8.19E-07	7.16E-01	0.00E+00

Table 5-8: St. Lucie Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
8	Bypass (Interfacing System LOCA)	1.43E+06	6.18E-08	8.87E-02	6.18E-08	8.87E-02	0.00E+00
CDF	All CET end states	N/A	6.93E-06	9.51E-01	6.93E-06	9.51E-01	8.68E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-9: St. Lucie Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	4.41E+03	3.99E-06	1.76E-02	3.99E-06	1.76E-02	-9.15E-07
2	Large Isolation Failures (Failure to Close)	1.14E+06	5.89E-08	6.72E-02	5.89E-08	6.72E-02	0.00E+00
3a	Small Isolation Failures (liner breach)	4.41E+04	2.24E-07	9.86E-03	2.24E-07	9.86E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	4.41E+05	5.59E-08	2.47E-02	5.61E-08	2.48E-02	9.15E-05
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 5-9: St. Lucie Unit 1 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
7	Failures Induced by Phenomena (Early and Late)	8.75E+05	5.86E-07	5.12E-01	5.86E-07	5.12E-01	0.00E+00
8	Bypass (Interfacing System LOCA)	1.43E+06	5.19E-07	7.45E-01	5.19E-07	7.45E-01	0.00E+00
CDF	All CET end states	N/A	5.44E-06	1.38E+00	5.44E-06	1.38E+00	9.06E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-10: St. Lucie Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years

Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	4.41E+03	5.59E-06	2.47E-02	5.59E-06	2.47E-02	-1.32E-06
2	Large Isolation Failures (Failure to Close)	1.14E+06	7.76E-08	8.86E-02	7.76E-08	8.86E-02	0.00E+00
3a	Small Isolation Failures (liner breach)	4.41E+04	3.13E-07	1.38E-02	3.13E-07	1.38E-02	0.00E+00
3b	Large Isolation Failures (liner breach)	4.41E+05	7.81E-08	3.45E-02	7.84E-08	3.46E-02	1.32E-04
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 5-10: St. Lucie Unit 2 Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person-Rem/yr (50 miles)**	
7	Failures Induced by Phenomena (Early and Late)	8.75E+05	8.19E-07	7.16E-01	8.19E-07	7.16E-01	0.00E+00
8	Bypass (Interfacing System LOCA)	1.43E+06	6.18E-08	8.87E-02	6.18E-08	8.87E-02	0.00E+00
CDF	All CET end states	N/A	6.93E-06	9.66E-01	6.93E-06	9.66E-01	1.30E-04

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 Step 4 - Determine the Change in Risk in Terms of Large Early Release Frequency (LERF)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the EPRI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For St. Lucie Unit 1 and Unit 2, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). For Unit 1, the baseline LERF based on a test frequency of three times in ten years is 1.12E-08/yr. Based on a ten year test interval from Table 5-11, the Class 3b frequency (conservatively including corrosion) is 3.74E-08/yr; and, based on a fifteen year test interval from Table 5-11, it is 5.61E-08/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from three to fifteen years for Unit 1 is 4.49E-08/yr as shown in Table 5-11. Similarly, the increase due to increasing the interval from ten to fifteen years is 1.87E-08/yr as shown in Table 5-11. For Unit 2, the baseline LERF based on a test frequency of three times in ten years is 1.57E-08/yr. Based on a ten year test interval from Table 5-12, the Class 3b frequency (conservatively including corrosion) is 5.22E-08/yr; and, based on a fifteen year test interval from Table 5-12, it is 7.84E-08/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from three to fifteen years for Unit 2 is 6.28E-08/yr as shown in Table 5-12. Similarly, the increase due to increasing the interval from ten to fifteen years is 2.62E-08/yr as shown in Table 5-12.

As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF for St. Lucie Unit 1 and Unit 2 is below the threshold criteria for a very small change when comparing both the fifteen year results to the current ten year requirement, and the fifteen year results compared to the original three year requirement. See Table 5-11 and Table 5-12 for more information.

5.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability (CCFP)

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis.

One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the EPRI Report No. 1009325, Revision 2-A. The NRC has previously accepted similar calculations (Reference 9) as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. The list below shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects.

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

$CCFP_{3,\ Unit\ 1}$ = [1 - (4.22E-06/yr + 4.47E-08/yr) / 5.44E-06/yr] * 100% = 21.61%

$CCFP_{3,\ Unit\ 1}$ = 21.61%

$CCFP_{10,\ Unit\ 1}$ = [1 - (4.09E-06/yr + 1.49E-07/yr) / 5.44E-06/yr] * 100% = 22.09%

$CCFP_{10,\ Unit\ 1}$ = 22.09%

$CCFP_{15,\ Unit\ 1}$ = [1 - (3.99E-06/yr + 2.24E-07/yr) / 5.44E-06/yr] * 100% = 22.43%

$CCFP_{15,\ Unit\ 1}$ = 22.43%

$\Delta CCFP_{Unit\ 1} = CCFP_{15,\ Unit\ 1} - CCFP_{3,\ Unit\ 1}$ = 0.83%

$\Delta CCFP_{Unit\ 1} = CCFP_{15,\ Unit\ 1} - CCFP_{10,\ Unit\ 1}$ = 0.34%

$\Delta CCFP_{Unit\ 1} = CCFP_{10,\ Unit\ 1} - CCFP_{3,\ Unit\ 1}$ = 0.48%

$CCFP_{3,\ Unit\ 2}$ = [1 - (5.90E-06/yr + 6.25E-08/yr) / 6.93E-06/yr] * 100% = 14.04%

$CCFP_{3,\ Unit\ 2}$ = 14.04%

$CCFP_{10,\ Unit\ 2}$ = [1 - (5.72E-06/yr + 2.08E-07/yr) / 6.93E-06/yr] * 100% = 14.57%

$CCFP_{10,\ Unit\ 2}$ = 14.57%

$CCFP_{15,\ Unit\ 2}$ = [1 - (5.59E-06/yr + 3.13E-07/yr) / 6.93E-06/yr] * 100% = 14.95%

$CCFP_{15,\ Unit\ 2}$ = 14.95%

$\Delta CCFP_{Unit\ 2} = CCFP_{15,\ Unit\ 2} - CCFP_{3,\ Unit\ 2}$ = 0.90%

$\Delta CCFP_{Unit\ 2} = CCFP_{15,\ Unit\ 2} - CCFP_{10,\ Unit\ 2}$ = 0.38%

$\Delta CCFP_{Unit\ 2} = CCFP_{10,\ Unit\ 2} - CCFP_{3,\ Unit\ 2}$ = 0.53%

The change in CCFP of approximately 0.83% for Unit 1 and 0.90% for Unit 2 by extending the test interval to fifteen years from the original three in ten year requirement is judged to be very small.

5.6 Summary of Results

The results from this ILRT extension risk assessment for St. Lucie Unit 1 and Unit 2 are summarized in Table 5-11 and Table 5-12.

Table 5-11: St. Lucie Unit 1 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	**Per-Rem/Yr**	**CDF/Yr**	**Per-Rem/Yr**	**CDF/Yr**	**Per-Rem/Yr**
1	4.41E+03	4.22E-06	1.86E-02	4.09E-06	1.80E-02	3.99E-06	1.76E-02
2	1.14E+06	5.89E-08	6.72E-02	5.89E-08	6.72E-02	5.89E-08	6.72E-02
3a	4.41E+04	4.47E-08	1.97E-03	1.49E-07	6.57E-03	2.24E-07	9.86E-03
3b	4.41E+05	1.12E-08	4.95E-03	3.74E-08	1.65E-02	5.61E-08	2.48E-02
7	8.75E+05	5.86E-07	5.12E-01	5.86E-07	5.12E-01	5.86E-07	5.12E-01
8	1.43E+06	5.19E-07	7.45E-01	5.19E-07	7.45E-01	5.19E-07	7.45E-01
Total	N/A	5.44E-06	1.35E+00	5.44E-06	1.37E+00	5.44E-06	1.38E+00
ILRT Dose Rate from 3a and 3b Per-Rem/Yr		6.92E-03		2.31E-02		3.46E-02	
Delta Total Dose Rate[1]	From 3 yr	N/A		1.56E-02		2.67E-02	
	From 10 yr	N/A		N/A		1.12E-02	
% change in dose rate from base	From 3 yr	N/A		1.15%		1.98%	
	From 10 yr	N/A		N/A		0.82%	
3b Frequency (LERF) Per-Rem/Yr		1.12E-08		3.74E-08		5.61E-08	
Delta LERF	From 3 yr	N/A		2.61E-08		4.49E-08	
	From 10 yr	N/A		N/A		1.87E-08	
CCFP %		21.61%		22.09%		22.43%	
Delta CCFP %	From 3 yr	N/A		0.48%		0.83%	
	From 10 yr	N/A		N/A		0.34%	

[1] The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

Table 5-12: St. Lucie Unit 2 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	4.41E+03	5.90E-06	2.60E-02	5.72E-06	2.52E-02	5.59E-06	2.47E-02
2	1.14E+06	7.76E-08	8.86E-02	7.76E-08	8.86E-02	7.76E-08	8.86E-02
3a	4.41E+04	6.25E-08	2.76E-03	2.08E-07	9.19E-03	3.13E-07	1.38E-02
3b	4.41E+05	1.57E-08	6.92E-03	5.22E-08	2.31E-02	7.84E-08	3.46E-02
7	8.75E+05	8.19E-07	7.16E-01	8.19E-07	7.16E-01	8.19E-07	7.16E-01
8	1.43E+06	6.18E-08	8.87E-02	6.18E-08	8.87E-02	6.18E-08	8.87E-02
Total	N/A	6.93E-06	9.29E-01	6.93E-06	9.51E-01	6.93E-06	9.66E-01
ILRT Dose Rate from 3a and 3b Per-Rem/Yr		9.68E-03		3.22E-02		4.84E-02	
Delta Total Dose Rate[1]	From 3 yr	N/A		2.18E-02		3.74E-02	
	From 10 yr	N/A		N/A		1.56E-02	
% change in dose rate from base	From 3 yr	N/A		2.34%		4.02%	
	From 10 yr	N/A		N/A		1.64%	
3b Frequency (LERF) Per-Rem/Yr		1.57E-08		5.22E-08		7.84E-08	
Delta LERF	From 3 yr	N/A		3.66E-08		6.28E-08	
	From 10 yr	N/A		N/A		2.62E-08	
CCFP %		14.04%		14.57%		14.95%	
Delta CCFP %	From 3 yr	N/A		0.53%		0.90%	
	From 10 yr	N/A		N/A		0.38%	

[1] The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

6 Sensitivities

6.1 Sensitivity to Corrosion Impact Assumptions

The St. Lucie Unit 1 and Unit 2 results in Table 5-5 through Table 5-10 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters 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 upper containment and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6-1. 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 increases in LERF due to corrosion of only 4.13E-12/yr for Unit 1 and 5.95E-12/yr for Unit 2. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6-1: Steel Plate Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Unit 1 Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 Years (per Rx-yr)		Unit 2 Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 Years (per Rx-yr)	
			Total Increase	Increase Due to Corrosion	Total Increase	Increase Due to Corrosion
Base Case Doubles every 5 yrs	Base Case (1% Upper Containment, 0.1% Basemat)	Base Case (10% Upper Containment, 100% Basemat)	4.49E-08	1.66E-10	6.28E-08	2.39E-10
Doubles every 2 yrs	Base	Base	4.50E-08	2.95E-10	6.29E-08	4.25E-10
Doubles every 10 yrs	Base	Base	4.48E-08	4.78E-11	6.26E-08	6.88E-11
Base	Base	15%	4.49E-08	2.32E-10	6.29E-08	3.34E-10
Base	Base	5%	4.48E-08	9.97E-11	6.27E-08	1.43E-10
Base	10% Upper Containment, 1% Basemat	Base	4.64E-08	1.66E-09	6.49E-08	2.39E-09
Base	0.1% Upper Containment, 0.01% Basemat	Base	4.47E-08	1.66E-11	6.25E-08	2.39E-11
Lower Bound						
Doubles every 10 yrs	0.1% Upper Containment, 0.01% Basemat	5% Upper Containment, 1% Basemat	4.47E-08	2.87E-15	6.25E-08	4.13E-15
Upper Bound						
Doubles every 2 yrs	10% Upper Containment, 1% Basemat	15% Upper Containment, 100% Basemat	4.47E-08	4.13E-12	6.25E-08	5.95E-12

6.2 Sensitivity to Class 3B Contribution to LERF

For Unit 1, the Class 3b frequency for the base case of a three in ten year ILRT interval including corrosion is 1.12E-08/yr (Table 5-5). Extending the interval to one in ten years results in a frequency of 3.74E-08/yr (Table 5-7). Extending it to one in fifteen years results in a frequency of 5.61E-08/yr (Table 5-9), which is an increase of 4.49E-08/yr from three in ten years to once in fifteen years.

For Unit 2, the Class 3b frequency for the base case of a three in ten year ILRT interval including corrosion is 1.57E-08/yr (Table 5-6). Extending the interval to one in ten years results in a frequency of 5.22E-08/yr (Table 5-8). Extending it to one in fifteen years results in a frequency of 7.84E-08/yr (Table 5-10), which is an increase of 6.28E-08/yr from three in ten years to once in fifteen years.

If 100% of the Class 3b sequences are assumed to have potential releases large enough for LERF, then the increase in LERF for Unit 1 and Unit 2 due to extending the interval from three in ten to one in fifteen is below the RG 1.174 threshold for very small changes in LERF of 1.00E-07/yr.

6.3 Potential Impact From External Events Contribution

The latest information related to external events for St. Lucie Unit 1 and Unit 2 is from the Extended Power Uprate submittals. These submittals included information which were extended from the Individual Plant Examination for External Events (IPEEE) submittals. The external events considered included fire, seismic, high winds, external flooding, and nearby facility and transportation accidents.

Fire Assessment

Early assessments of fire risk vulnerabilities were performed as part of the IPEEE. St. Lucie analyses utilized the EPRI Five Induced Vulnerability Evaluation (FIVE) process. Using that methodology a bounding fire induced CDF exceeded 1.0E-04 per year. Since that time various fire-related plant improvements have been made, fire assessments have been refined and updated regulatory applications including fire re-assessments have been performed. In 2010, in support the EPU assessments (References 25 and 26) fire risks were re-evaluated. Results of these assessments for St. Lucie Unit 1 indicated that a realistic assessment of fire-related core damage frequencies (CDFs) are less than 5.0E-06 per year and LERF contributions less than 5.0E-07 per year. Similar evaluations performed for St. Lucie 2 indicated a fire CDF contribution of 9.2E-06 per year and a LERF contribution of less than 9.2E-07 per year. A more recent and conservative assessment of fire risk following NUREG/CR-6850 in support of an NFPA-805 application identifies St. Lucie Unit 1 and Unit 2 fire CDF as 5.12E-05/yr and 6.70E-05/yr, respectively. Similarly, fire induced LERF were quantified for NFPA-805 application as 7.00E-06/yr (Unit1) and 7.90E-06/yr (Unit 2) (Reference 29).

Seismic Assessment

The St. Lucie units are located in a zone of low seismicity. Consequently, the IPEEE seismic assessment implemented a limited scope seismic screening which consisted of reduced scope seismic walkdowns done in accordance with the plant's GL 87-02 walkdown procedure. As a result of the low seismic risk impact, a full seismic risk analysis has not been performed either during the license renewal or for EPU implementation. However, in order to provide additional insight with respect to the effect of the impact of EPU on seismic risk, a reduced seismic risk estimate was performed. Those evaluations showed seismic risks below a ground acceleration of 0.1g was negligible and that the impact of EPU on seismic risk was insignificant. In the NRC SER the NRC staff noted "Based on a simplified approach to estimate the core damage frequency from a seismic margins approach and using the latest published United States Geological Survey seismic hazards information, the staff estimates the St. Lucie seismic CDF is about or below 5.0E-05 per year." This assessment essentially assumes core damage occurs a little above the plant design g-level.

For the purposes of the ILRT extension effort, a simplified but site specific assessment seismic induced CDF estimate is developed. The process used involved the generation of a plant level fragility curve based on the 0.1g High Confidence of Low Probability of Failure (HCLPF) value for the St. Lucie plants as documented in Reference 32 and assuming a composite variability β_C of 0.4. The value of β_C is consistent with the GI-199 safety/risk evaluation performed by the NRC. As the HCLPF capacity is approximately defined as 1% conditional probability of failure, the relationship between HCLPF and the median capacity (Am) is described by the following equation (see Reference 30, equation 4-13):

$$\mathrm{HCLPF} = A_m e^{-2.33\beta_C} \qquad (1)$$

Using the above assumptions A_m can be estimated to be approximately 0.25 g. The median capacity and the composite variability can be then used to generate a fragility curve, using the following equation (2) (see Reference 30, equation 4-2).

$$p(a) = \Phi\left(\frac{\ln\left(\frac{a}{A_m}\right)}{\beta_C}\right) \qquad (2)$$

In this equation Φ is the standard Gaussian cumulative distribution function, and a is the ground acceleration. Equation (2) therefore yields the estimated plant level fragility curve.

Once the plant level fragility curve has been estimated, it can then be integrated with the mean hazard curve for the site. The most updated seismic hazard estimate available for the St. Lucie plant was generated by EPRI within the framework of the Post Fukushima Recommendation 2.1 and uses the 2012 Central and East United States (CEUS) Seismic Source Characterization (SSC) documented in NUREG 2115 (Reference 33) and the 2013 Ground Motion Model (GMM) update generated by EPRI (Reference 34).

The hazard estimate calculated for St. Lucie is documented in Reference 35.

Integrating the plant level fragility curve obtained with the above seismic capacity parameters for the St. Lucie plant with the most updated seismic hazard information, the estimated Seismic CDF is approximately 3.49E-06 per year. Under the reasonable expectation that the plant HCLPF is not driven by significant structural failures, it can be assumed that early large releases will occur at a rate of 10% of the CDF, a LERF estimate would be on the order of 3.49E-07 per year.

Note, a limited scope seismic margin estimate for St. Lucie performed in support to GI-199 indicates that the plant HCLPF documented in the IPEEE and used in the above assessment is extremely conservative and that the actual plant HCLPF can be as high as 0.3g. The limited scope margin assessment is nevertheless not credited for the ILRT evaluation.

Other External Hazards

An evaluation of high winds, external floods and transportation events and other hazards were performed during the IPEEE following the progressive screening methodology in NUREG-1407. This included a review of the external event hazards at the plant and the licensing basis, an assessment of whether there have been any significant changes since the Operating License was issued, a site walkdown (where appropriate), and screening to determine if the plant meets the 1975 Standard Review Plan (SRP) criteria. The results indicated there are no significant changes to the site since the Operating Licensing was issued that would alter the assessment of these hazard impacts, and that the plant does comply with the 1975 SRP criteria. As these hazards were screen during the IPEEE, no core damage or LERF frequencies were developed at that time.

To support the EPU an upper bound estimate of other hazards was established assuming these external events are likely to result in a LOOP where power recovery in the short term is unlikely. Consistent with the EPU assessment it is assumed that the frequency of these external hazards are bounded by weather induced LOOP (frequency of 0.00528 per year) with offsite power not recoverable for an indefinite period. While recoverable weather induced LOOPs have been observed with some frequency, unrecoverable events are rare. Thus, use of weather induced initiating event frequency to bound the non-recovered external event frequency is considered appropriate. Results of these assessments are presented in table 6-2.

Table 6-2: Estimated CDF and LERF Contribution for Other Hazards

Plant	CDF/yr	LERF/yr
St. Lucie Unit 1	1.08E-06	1.87E-07
St. Lucie Unit 2	1.72E-07	3.41E-08

External Events Summary

Table 6-3 below lists the St. Lucie CDF values for each external event type that are used to determine the potential impact from the External Events contribution.

Table 6-3: St. Lucie External Events Summary

External Event Type	St. Lucie Unit 1		St. Lucie Unit 2	
	CDF/yr	**LERF/yr**	**CDF/yr**	**LERF/yr**
Internal Events	5.44E-06	7.79E-07	6.93E-06	2.32E-07
Seismic	3.49E-06	3.49E-07	3.49E-06	3.49E-07
Other Hazards (High Winds /External Floods/Transportation)	1.08E-06	1.87E-07	1.72E-07	3.41E-08
Fire	5.12E-05	7.00E-06	6.70E-05	7.90E-06
Total	6.12E-05	8.32E-06	7.76E-05	8.52E-06

Combining the External Events CDF values and the Internal Events CDF yields a CDF estimate of 6.12E-05/yr (Unit 1) and 7.76E-05/yr (Unit 2). LERF estimates including External Events are 8.32E-06/yr (Unit 1) and 8.52E-06/yr (Unit 2).

The change in LERF from extending the Type A test interval can be conservatively estimated using the total CDF values to determine the external event contribution. These CDF values were specifically used to determine the Class 3b frequency (neglecting corrosion[5]) based on the external events contribution. The factors for determining the increase in the non-detection probability of a leak described in Section 4.3 were applied to the Class 3b base value frequencies to determine the 3b frequencies for the once per ten year test and once per fifteen year test for each unit.

Class 3b Frequency (three per ten year test) = 0.0023 * (CDF - LERF)

Class 3b Frequency (Unit 1) = 0.0023 * (6.12E-05/yr – 8.32E-06/yr) = 1.22E-07/yr

Class 3b Frequency (Unit 1) (once per ten year test) = 3.33 * 1.22E-07/yr = 4.05E-07/yr

Class 3b Frequency (Unit 1) (once per fifteen year test) = 5.00 * 1.22E-07/yr = 6.08E-07/yr

Class 3b Frequency (Unit 2) = 0.0023 * (7.76E-05/yr – 8.52E-06/yr) = 1.59E-07/yr

Class 3b Frequency (Unit 2) (once per ten year test) = 3.33 * 1.59E-07/yr = 5.29E-07/yr

[5] Corrosion effects are not explicitly considered in the sensitivity assessment as the impact is negligible.

Class 3b Frequency (Unit 2) (once per fifteen year test) = 5.00 * 1.59E-07/yr = 7.95E-07/yr

Table 6-4 shows the results of these calculations. Note that in the above calculation Class 3 b releases are considered to arise from a change in state of prior non-LERF states to a LERF (Class 3b) state.

Table 6-4: St. Lucie Estimated Total LERF Including External Events Impact				
Case	**3b Frequency (3 per 10 year test) per year**	**3b Frequency (1 per 10 year test) per year**	**3b Frequency (1 per 15 year test) per year**	**LERF Increase (3 per 10 to 1 per 15 year test) per year**
Unit 1 Internal Events Contribution (From Base Case Table 5-11)	1.12E-08	3.72E-08	5.58E-08	4.47E-08
Unit 1 Total Contribution including External Events	1.22E-07	4.05E-07	6.08E-07	4.87E-07
Unit 2 Internal Events Contribution (From Base Case Table 5-12)	1.57E-08	5.22E-08	7.84E-08	6.27E-08
Unit 2 Total Contribution including External Events	1.59E-07	5.29E-07	7.95E-07	6.36E-07

Using the above approach results in a total LERF (Class 3b) value of 6.08E-07/yr for a permanent once per 15 year ILRT program for Unit 1 and 7.95E-07/yr for Unit 2. These frequencies remain below the Regulatory Guide 1.174 criteria of 1.00E-05/yr following the ILRT extension. Furthermore, the increase in total LERF from the three per ten year test to the once per fifteen year test is 4.87E-07/yr for Unit 1 and 6.36E-07/yr for Unit 2, both of which are within the range of the Regulatory Guide 1.174 criteria of 1.00E-07/yr to 1.00E-06/yr for a small change in risk.

6.4 Sensitivity to Release Class Dose

The present assessment focuses on the use of the Reference 1 methodology with population doses based on extrapolation of Surry dose estimates to the St. Lucie site. This sensitivity assesses the impact of the use of St. Lucie site specific doses. The release class-population dose mapping use in this assessment is based on use of results of site specific MAAP and MACCS analyses performed in support of license renewal assessment of Severe Accident Management Alternatives (SAMA). This information was utilized in the one-time 16 yr ILRT extension (Reference 27). In this assessment the dose estimates used in that submittal for the various release classes are adjusted for the current St. Lucie power level and anticipated increase in population from 2025 (used in the license renewal submittal, Reference 31) to 2040. With the exception of the release class bin dose rates, all other calculations used in the assessment are unchanged. Results of this evaluation are presented in Table 6-5 (St. Lucie Unit 1) and Table 6-6 (St. Lucie Unit 2).

Table 6-5: Sensitivity to Release Class Dose for St. Lucie Unit 1 ILRT Cases

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	7.86E+03	4.22E-06	3.32E-02	4.09E-06	3.21E-02	3.99E-06	3.14E-02
2	7.34E+06	5.89E-08	4.32E-01	5.89E-08	4.32E-01	5.89E-08	4.32E-01
3a	7.86E+04	4.47E-08	3.51E-03	1.49E-07	1.17E-02	2.24E-07	1.76E-02
3b	7.86E+05	1.12E-08	8.82E-03	3.74E-08	2.94E-02	5.61E-08	4.41E-02
7	1.20E+07	5.86E-07	7.01E+00	5.86E-07	7.01E+00	5.86E-07	7.01E+00
8	7.22E+06	5.19E-07	3.75E+00	5.19E-07	3.75E+00	5.19E-07	3.75E+00
Total	N/A	5.44E-06	1.12E+01	5.44E-06	1.13E+01	5.44E-06	1.13E+01
ILRT Dose Rate from 3a and 3b		1.23E-02		4.11E-02		6.17E-02	
Delta Total Dose Rate	N/A	N/A		2.77E-02		4.76E-02	
	N/A	N/A		N/A		1.99E-02	
% change in dose rate from base	From 3 yr	N/A		0.25%		0.42%	
	From 10 yr	N/A		N/A		0.18%	
3b Frequency (LERF)		1.12E-08		3.74E-08		5.61E-08	
Delta LERF	From 3 yr	N/A		2.61E-08		4.49E-08	
	From 10 yr	N/A		N/A		1.87E-08	
CCFP %		21.61%		22.09%		22.43%	
Delta CCFP%	From 3 yr	N/A		0.48%		0.83%	
	From 10 yr	N/A		N/A		0.34%	

Table 6-6: Sensitivity to Release Class Dose for St. Lucie Unit 2 ILRT Cases

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	7.86E+03	5.90E-06	4.64E-02	5.72E-06	4.49E-02	5.59E-06	4.39E-02
2	7.34E+06	7.76E-08	5.70E-01	7.76E-08	5.70E-01	7.76E-08	5.70E-01
3a	7.86E+04	6.25E-08	4.91E-03	2.08E-07	1.64E-02	3.13E-07	2.46E-02
3b	7.86E+05	1.57E-08	1.23E-02	5.22E-08	4.11E-02	7.84E-08	6.17E-02
7	1.20E+07	8.19E-07	9.80E+00	8.19E-07	9.80E+00	8.19E-07	9.80E+00
8	7.22E+06	6.18E-08	4.46E-01	6.18E-08	4.46E-01	6.18E-08	4.46E-01
Total	N/A	6.93E-06	1.09E+01	6.93E-06	1.09E+01	6.93E-06	1.09E+01
ILRT Dose Rate from 3a and 3b		1.72E-02		5.74E-02		8.62E-02	
Delta Total Dose Rate	From 3 yr	N/A		3.88E-02		6.65E-02	
	From 10 yr	N/A		N/A		2.78E-02	
% change in dose rate from base	From 3 yr	N/A		0.36%		0.61%	
	From 10 yr	N/A		N/A		0.25%	
3b Frequency (LERF)		1.57E-08		5.22E-08		7.84E-08	
Delta LERF	From 3 yr	N/A		3.66E-08		6.28E-08	
	From 10 yr	N/A		N/A		2.62E-08	
CCFP %		14.04%		14.57%		14.95%	
Delta CCFP%	From 3 yr	N/A		0.53%		0.91%	
	From 10 yr	N/A		N/A		0.38%	

Note that use of site specific dose estimates results in a marginal increase in ILRT doses (2.67E-02 to 4.76E-02 Person-Rem/year, Unit 1, and 3.74E-02 to 6.65E-02 Person-Rem/year, Unit 2) and a reduction in the percent change in doses from several percent to under 1% for each unit.

7 Conclusions

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the following conclusions regarding the assessment of the plant risk are associated with permanently extending the Type A ILRT test frequency to once in fifteen years:

- Regulatory Guide 1.174 (Reference 4) provides guidance for determining the risk impact of plant specific changes to the licensing basis. Regulatory 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. The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is conservatively estimated as 4.49E-08/yr for Unit 1 and 6.28E-08/yr for Unit 2 using the EPRI guidance as written. As such, the estimated change in LERF for Unit 1 and Unit 2 is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

- Regulatory Guide 1.174 (Reference 4) also states that when the calculated increase in LERF is in the range of 1.00E-07 per reactor year to 1.00E-06 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.00E-05 per reactor year. An additional assessment of the impact from External Events was also made. In this case, the total class 3b contribution to LERF including External Events was conservatively estimated as 6.08E-07/yr for St. Lucie Unit 1 and 7.95E-07/yr for St. Lucie Unit 2. This is below the RG 1.174 acceptance criteria for total LERF of 1.00E-05/yr and therefore this change satisfied both the incremental and absolute expectations with regard to the RG 1.174 LERF metric.

- The change in Type A test frequency to once per fifteen years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 2.67E-02 person-rem/yr for Unit 1 and 3.74E-02 person-rem/yr for Unit 2. Note that this value is based on internal events only and does not consider external events. EPRI Report No. 1009325, Revision 2-A 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. This is consistent with the NRC final safety evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 24). Moreover, the risk impact when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure probability from the three in ten year interval to a permanent one time in fifteen year interval is 0.83% for Unit 1 and 0.90% for Unit 2. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of ≤1.5 percentage points are very small. This is consistent with the NRC final safety evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 24). Therefore this increase is judged to be very small.

Therefore, permanently increasing the ILRT interval to fifteen years is considered to be a very small change to the St. Lucie Unit 1 and Unit 2 risk profile.

7.1.1 Previous Assessments

The NRC in NUREG-1493 (Reference 6) has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per ten years to one 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 Type 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, increasing the interval between integrated leakage rate tests is possible with minimal impact on public risk. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for St. Lucie Unit 1 and Unit 2 confirm these general findings on a plant specific basis considering the severe accidents evaluated for St. Lucie Unit 1 and Unit 2, the St. Lucie Unit 1 and Unit 2 containment failure modes, and the local population surrounding St. Lucie.

8 References

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

2. Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.

3. 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.

4. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis, Regulatory Guide 1.174, Revision 2, May 2011.

5. 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. Performance-Based Containment Leak-Test Program, NUREG-1493, September 1995.

7. Evaluation of Severe Accident Risks: Surry Unit 1, Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, December 1990.

8. Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.

9. United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

10. Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.

11. Reliability Analysis of Containment Isolation Systems, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.

12. Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check', NUREG-1273, April 1988.

13. Review of Light Water Reactor Regulatory Requirements, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.

14. Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA TR-105189, Final Report, May 1995.

15. Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, December 1990.

16. United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.

17. Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.

18. Letter from J.A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.

19. Letter from D.E. Young (Florida Power, Crystal River) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.

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

21. Letter from P.P. Sena III (FENOC) to Document Control Desk (NRC), dated June 18, 2009, Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, LER 2009-003-00, "Containment Liner Through Wall Defect Due to Corrosion."

22. Letter from J.E. Pollock (AEP Indiana Michigan Power) to Document Control Desk (NRC), dated March 16, 2001, submitting LER 316/2000-001-01, "Through-Liner Hole Discovered in Containment Liner."

23. St. Lucie Units 1 and 2, Docket No. 50-335 and 50-389 NRC Generic Letter 88-20 Supplement 4 Individual Plant Examinations of External Events for Severe Accidents Vulnerabilities Report, December 15, 1994.

24. Final Safety Evaluation for NEI Topical Report 94-01 Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" and EPRI Report No. 1009325 Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Test Intervals".

25. Letter Orf, T.J. (NRC) to M. Nazar (FP&L)," St. Lucie Plant Unit 1-Issuance of Amendment No. 213 Regarding Extended Power Uprate," July 9, 2012.

26. Letter Orf, T.J. (NRC) to M. Nazar (FP&L)," St. Lucie Plant Unit 2-Issuance of Amendment No. 163 Regarding Extended Power Uprate," September 24, 2012.

27. PSL-BFJR-02-004, Revision 2, "St. Lucie Units 1 and 2 ILRT Extension", FP&L, December 28, 2004.

28. PSL-ENG-SEOS-02-061, Revision 0, "Response to Requests for Additional Information Concerning the Proposed License Amendment for ILRT Interval Extension."

29. St. Lucie Nuclear Plant Fire PRA Summary Report NUREG/CR-6850 Task 16, Revision 4, March 2013.

30. Seismic Probabilistic Risk Assessment Implementation Guide, EPRI, Palo Alto, CA: 2013.

31. NUREG-1437 Supplement 11, "Generic Environment Impact Statement for License Renewal of Nuclear Plants", May 2003.

32. Generic Issue 199 (GI-199), Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants Safety/Risk Assessment, August 2012, Appendix C (ADAMS access number ML100270731).

33. NRC, EPRI, DOE (2012). "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," NRC Report NUREG-2115, EPRI Report 1021097, 6 Volumes.

34. EPRI (2013), EPRI (2004, 2006), "Ground-Motion Model (GMM) Review Project," Electric Power Research Institute, Palo Alto, CA, Rept. 3002000717, June, 2 Volumes.

35. EPRI letter RSM-012414-057, "St. Lucie Soil Seismic Hazard and Screening Report," February 14, 2014.

Attachment 6

Summary of St. Lucie LLRT Results Not Demonstrating Acceptable Performance

PSL-1 LLRT Failures 2005-2014

Component	Date	Leakage (SCCM)	Limit (SCCM)	A-F/A-L	Penetration	AR / Ref #
Door Seal	10/30/13	*	9080	A-L	Pers Airlock	01916343
FCV-25-4	10/17/13	180000	50000	A-F	P-10	01911038
SE-01-1	10/15/13	3600	2700	A-F	P-44	01912378
FCV-25-5	10/10/13	*	50000	A-F	P-10	01911038
FCV-26-3	12/3/11	23500	3620	A-F	P-52B	01712224
FCV-26-3	4/10/10	4500	3620	A-F	P-52B	00474657
V5204	11/13/08	4400	2000	A-L	P-29A	00455527
SE-01-1	11/12/08	3400	2700	A-F	P-44	00515736
FCV-25-5	11/1/08	145000	133000	A-F	P-10	00455223
FCV-25-4	4/16/07	*	133000	A-F	P-10	00482832
V18195	4/15/07	*	14500	A-F	P-9	00479056
FCV-25-7	11/29/05	90000	66680	A-L	P-67	00424986 (00424819)
V2516	11/29/05	135000	7250	A-L	P-26	00480680
V5204	11/24/05	65000	2000	A-L	P-29A	00480526
V5204	10/30/05	2000	2000	A-F	P-29A	00424762
MV-18-1	10/21/05	8200	7250	A-F	P-9	00423709
FCV-25-2	10/21/05	130000	133000	A-F	P-11	00423699

PSL-2 LLRT Failures 2005-2014

Component	Date	Leakage (SCCM)	Limit (SCCM)	A-F/A-L	Penetration	AR / Ref #
FCV-25-5	4/3/14	*	48769	A-L	P-10	01954343
V5201	3/23/14	3000	2000	A-L	P-29A	01951044
SE-03-2A/B	3/10/14	8600	5796	A-F	P-41	01947012
FCV-25-36	10/28/12	10200	8000	A-F	P-56	01817385
SH18797	8/21/12	9000	2898	A-F	P-8	01795943
V6792	8/18/12	5400	2000	A-F	P-14	01795130
SE-03-2B	8/16/12	25000	5796	A-F	P-41	01794724 (40063405)
SE-05-1A	2/19/11	4800	2000	A-F	P-28A	01621741
SE-03-2A	1/17/11	6600	5796	A-F	P-41	01609506 (40063404)
SE-05-1C	9/12/09	10200	2000	A-F	P-28A	00464458
FCV-25-20	5/12/09	60000	47250	A-L	P-57	00465084
HCV-18-2	5/4/09	5400	2898	A-F	P-8	00460398
V5201	11/30/07	50000	2000	A-L	P-29A	00510386
FCV-25-36	11/24/07	23500	8000	A-L	P-56	00484072
SE-03-2B	10/21/07	23500	5796	A-L	P-41	00509403 (37023274)
FCV-25-36	8/15/06	8400	8000	A-F	P-56	00481633
SE-05-1E	5/16/06	2400	2000	A-L	P-28A	00405119
HCV-18-2	5/2/06	40000	2898	A-F	P-8	00429461
V-25-20	4/28/06	150000	100000	A-F	P-67	00405027
FCV-25-3	2/5/05	180000	47250	A-L	P-11	00414936

*Could not pressurize penetration to test pressure.