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

August 27, 2015

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
NextEra Energy
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING CHANGE TO TECHNICAL SPECIFICATION 6.8.4.h, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (TAC NOS. MF4694 AND MF4695)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment Nos. 226 and 176 to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2, respectively. These amendments consist of changes to the technical specifications (TSs) in response to your application dated August 26, 2014, as supplemented by letters dated January 14, February 6, and May 14, 2015.

Each amendment revises TS 6.8.4.h, "Containment Leakage Rate Testing Program," to allow extension of the 10-year frequency of the Type A test (i.e., integrated leak rate test) to a 15-year frequency. The proposed change is accomplished by replacing the previous wording, which references Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," with wording referencing Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [*Code of Federal Regulations*] Part 50, Appendix J," October 2008.

The NRC staff's related safety evaluation of the amendments is enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50-389

Enclosures:
1. Amendment No. 226 to DPR-67
2. Amendment No. 176 to NPF-16
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 226
Renewed License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company (FPL, the licensee), dated August 26, 2014, as supplemented by letters dated January 14, February 6, and May 14, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, Renewed Facility Operating License No. DPR-67 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 3.B to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.226 , are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 27, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 226

RENEWED FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace page 3 of the Renewed Facility Operating License No. DPR-67 with the revised page 3.

Replace the following page of Appendix A, Technical Specifications, with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 3020 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 226 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

Appendix B, the Environmental Protection Plan (Non-Radiological), contains environmental conditions of the renewed license. If significant detrimental effects or evidence of irreversible damage are detected by the monitoring programs required by Appendix B of this license, FPL will provide the Commission with an analysis of the problem and plan of action to be taken subject to Commission approval to eliminate or significantly reduce the detrimental effects or damage.

C. Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 28, 2003, describes certain future activities to be completed before the period of extended operation. FPL shall complete these activities no later than March 1, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on March 28, 2003, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed license. Until that update is complete, FPL may make changes to the programs described in such supplement without prior Commission approval, provided that FPL evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Sustained Core Uncovery Actions

Procedural guidance shall be in place to instruct operators to implement actions that are designed to mitigate a small-break loss-of-coolant accident prior to a calculated time of sustained core uncovery.

(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 $\geq$ 10 psig.

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FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 176
Renewed License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company (FPL, the licensee), dated August 26, 2014, as supplemented by letters dated January 14, February 6, and May 14, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 2

2. Accordingly, Renewed Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 3.B to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 176 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 27, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 176

RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace page 3 of the Renewed Facility Operating License No. DPR-67 with the revised page 3.

Replace the following page of Appendix A, Technical Specifications, with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
6-15b	6-15b

neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required.

D. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission's regulations: 10 CFR Part 20, Section 30.34 of 10 FR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 3020 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 176 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 226 AND 176

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-67 AND NPF-16

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By letter dated August 26, 2014 (Reference 1), as supplemented by letters dated January 14, February 6, and May 14, 2015 (References 2, 3, and 4), Florida Power & Light Company (FPL, the licensee) submitted a license amendment request (LAR), proposing to change the Technical Specifications (TSs) for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie or SL-1 and 2). Specifically, the licensee proposed to revise TS 6.8.4.h, "Containment Leakage Rate Testing Program" (hereinafter "TS 6.8.4.h") to allow extension of the 10-year frequency of the Type A test (i.e., integrated leak rate test (ILRT)) to a 15-year frequency. The proposed change will be accomplished by replacing the current wording of Specification 6.8.4.h, which references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program" (Reference 8) with wording referencing Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," October 2008 (Reference 5).

The proposed amendment is consistent with the St. Lucie performance-based leakage testing programs based on Option B of 10 CFR Part 50, Appendix J. The licensee had previously submitted an LAR for St. Lucie to extend the ILRT intervals on a one-time basis from 10 years to 15 years (Reference 6). This one-time extension was approved by the U.S. Nuclear Regulatory Commission (NRC) as Amendment Nos. 187 and 130 for the SL-1 and 2 on April 10, 2003 (Reference 7).

The purpose of the August 26, 2014, LAR is to extend, on a permanent basis, the 10-year frequency of the Type A primary containment ILRT intervals to 15 years for both units. The scope of the proposed amendment was reduced with the supplement of January 14, 2015. In particular, FPL no longer sought an extension of the Type C local leak rate test (LLRT) from a 60-month frequency to a 75-month frequency, per the guidance of NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012 (Reference 15).

The licensee's supplements dated January 14, February 6, and May 14, 2015, provided additional information to support the original August 26, 2014, application. The supplements do not expand the scope of the original application and do not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 20, 2015 (80 FR 2750).

2.0 REGULATORY EVALUATION

Section 50.54(o) of Title 10 of the Code of *Federal Regulations* (10 CFR) requires that the primary reactor containments for water-cooled power reactors be subject to the requirements set forth in Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Two options are included in 10 CFR Part 50, Appendix J: "Option A – Prescriptive Requirements," and "Option B – Performance-Based Requirements," either of which can be chosen for meeting the requirements of the appendix. The testing requirements in 10 CFR Part 50, Appendix J, ensure that (a) leakage through containments or systems and components penetrating containments do not exceed allowable leakage rates specified in the TSs, and (b) integrity of the containment structure is maintained during the service life of the containment. The St. Lucie licensing bases have adopted and have been implementing Option B.

Option B of 10 CFR Part 50, Appendix J, specifies performance-based requirements and criteria for preoperational and subsequent leakage-rate testing. These requirements are met by performance of Type A tests to measure the containment system overall integrated leakage rate; Type B pneumatic tests to detect and measure local leakage rates across pressure-retaining leakage-limiting boundaries such as penetrations; and Type C pneumatic tests to measure containment isolation valve leakage rates. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each boundary and isolation valve (for Types B and C tests) to ensure integrity of the overall containment system as a barrier to fission product release.

LAR Enclosure, Section 3.3, "Supplemental Inspections," describes the SL-1 and 2 containments as follows:

> 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.

Currently, TS 6.8.4.h requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in RG 1.163. This RG endorses, with certain exceptions, Topical Report (TR) NEI 94-01, Revision 0 (Reference 9).

Guidance for extending Type A ILRT surveillance intervals beyond 10 years is provided in TR NEI 94-01, Revision 2, August 2007 (Reference 10). The NRC staff approved TR NEI 94-01 on June 25, 2008, subject to certain limitations and conditions (Reference 11).

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

Paragraph (a)(1) of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," states, in part, that the licensee:

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

A Type A test is an overall ILRT of the containment structure. NEI 94-01, Revision 0, specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months, but this "should be used only in cases where refueling schedules have been changed to accommodate other factors." With the incorporation of Amendment Nos. 187 and 130 into the SL-1 and 2 TSs, the long-term ILRT test interval requirement for SL-1 and 2 remains at 10 years.

The two most recent SL-1 Type A test results of May 1993 and December 2005 are captured in Section 3.1 of the LAR. Both Type A tests were performed consistent with the definition of P_a (peak calculated containment internal pressure for design basis LOCA). Both Type A tests were successful in that the test results were less than 1.0 L_a (maximum allowed containment leakage rate) and less than the SL-1 TS 6.8.4.h.a limiting values.

The two most recent SL-2 Type A test results of June 1992 and December 2007 are captured in Section 3.1 of the LAR. Both Type A tests were performed consistent with the definition of $P_{a,}$ and both were successful in that the test results were less than $1.0L_a$ and less than the SL-2 TS 6.8.4.h.a limiting values.

The Type A, Type B, and Type C test results must not exceed the L_a with margin, as specified in TS 6.8.4.h. Option B also requires that a general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration, which may affect the containment leak-tight integrity, must be conducted prior to each Type A test and at a periodic interval between tests based on the performance of the containment system.

FPL proposes to extend the SL-1 and 2 intervals for the primary containment ILRT to no longer than 15 years from the last ILRT. The last SL-1 ILRT was completed on December 8, 2005. The last SL-2 ILRT was completed on December 18, 2007. The ILRTs for each unit are currently required to be performed at a frequency of once every 10 years. Therefore, the next SL-1 ILRT is due during December 2015 and the next SL-2 ILRT is due during December 2017. Accordingly, the next ILRT for SL-1 would need to be performed during the spring 2015 refueling outage, and the next ILRT for SL-2 would need to be performed during the spring 2017 refueling outage. Using the proposed interval of no longer than 15 years, the next SL-1 ILRT will be due in December 2020, and the next SL-2 ILRT will be due in December 2022.

As 10 CFR 50, Appendix J, Option B, Section V.B.3, requires that the RG or other implementation document used by a licensee to develop a performance-based leakage-testing program must be included by general reference in the TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a RG.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Changes

The licensee proposed to revise a portion of TS 6.8.4.h for SL-1 as follows (new wording presented in **bold**):

> 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-I or American Nuclear Society (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.~~
>
> **NEI 94-01, Revision 2-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.**

The licensee proposed to revise TS 6.8.4.h for SL-2 as follows (new wording presented in **bold**):

> 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~~ **NEI 94-01, Revision 2-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 ~~allow~~ **allowed** containment leakage rate, L_a, at P_a, shall be 0.50 percent of containment air weight per day.

3.2 Historical Test and Surveillance Results

Consistent with the guidance in NEI 94-01, Revision 2-A, the licensee provided justification for the proposed change from the perspective of historical leakage testing results.

3.2.1 Previous Type A ILRT Results

Per the application, the licensee proposed to extend for both SL-1 and 2 the current performance-based Type A test intervals to 15 years by adopting NEI 94-01, Revision 2-A (Reference 5) as the implementation document in TS 6.8.4.h. This change would allow SL-1 to conduct the next Type A test by December 2020, in lieu of the current scheduled outage the spring of 2015. This change would allow SL-2 to conduct the next Type A test by December 2022, in lieu of the current scheduled outage the spring of 2017. The licensee justified these proposed changes by demonstrating adequate performance of the SL-1 and 2 containments based on plant-specific containment leakage testing program results and containment ISI (i.e., American Society of Mechanical Engineers (ASME) Subsection IWE) results.

The licensee indicated in Section 3.1 of the application, "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 SL-2, all with satisfactory results."

NEI 94-01, Revision 2-A, Section 9.1.2, provides that further extensions in test intervals are contingent upon two consecutive successful periodic Type A tests and the guidance as stated therein in Section 9.2.3.

For the two most recent SL-1 ILRTs:

- During the periodic Type A (ILRT) test of December 2005, the performance leak rate corresponding to the definition in NEI 94-01 was 0.2102 percent weight/day; and
- During the periodic Type A (ILRT) test of May 1993, the performance leak rate corresponding to the definition in NEI 94-01 was 0.133 percent weight/day.

For the two most recent SL-2 ILRTs:

- During the periodic Type A (ILRT) test of December 2007, the performance leak rate corresponding to the definition in NEI 94-01 was 0.1930 percent weight/day; and

- During the periodic Type A (ILRT) test of June 1992, the performance leak rate corresponding to the definition in NEI 94-01 was 0.052 percent weight/day.

TS 6.8.4.h.a establishes the maximum limit for SL-1 or 2 startup following completion of Type A testing at $\leq 0.75\ L_a$, which equals 0.375 percent of containment air weight per day. The SL-1 or 2 containments were designed for a leakage rate L_a not to exceed 0.5 percent by weight of containment air per 24 hours at the calculated peak pressure, P_a. Therefore, there has been substantial margin to the performance limit as described in TS 6.8.4.h of L_a equal to 0.5 percent weight/day.

The NRC staff notes that the last sentence of NEI 94-01, Revision 2-A, Section 9.2.3, "Extended Test Intervals," reads, "In the event where previous Type A tests were performed at reduced pressure (as described in 10 CFR 50, Appendix J, Option A), at least one of the two consecutive periodic Type A tests shall be performed at peak accident pressure (P_a)." Section 3.2 of the licensee's application, "Type B and Type C Testing (LLRT) Program," reads in part, "... To permit operation at Extended Power Uprate (EPU) conditions, the NRC issued Amendment No. 213 [Agencywide Documents Access and Management System (ADAMS) Accession No.] (ML12156A208), which raised the Unit 1 P_a value from 39.6 pounds per square inch gauge (psig) to 42.8 psig, and Amendment No. 163 (ADAMS Accession No. ML12235A463), which raised the Unit 2 P_a value from 41.8 psig to 43.5 psig. ..." Amendment No. 213 for SL-1 was issued on July 9, 2012. Amendment No. 163 for SL-2 was issued on September 24, 2012. Based on Section 9.2.3 and the timing of the EPU amendments, the NRC staff requested additional information from FPL to confirm that at least one of the actual ILRT test pressures employed during the two most recent Type A tests (per the guidance of American Nuclear Society (ANS) document 56.8-1994), bound the revised P_a values of Unit No. 1 Amendment No. 213 and Unit No. 2 Amendment No. 163, respectively.

The licensee responded by its May 14, 2015, letter, listing the actual Type A test pressures recorded at the start of and end of the ILRTs of December 2005 for SL-1 and December 2007 for SL-2. From this data, the NRC staff was able to confirm that the most recent ILRTs performed on SL-1 and 2 satisfied the requirements for Type A test pressures and bound the EPU-revised P_a values of Unit No. 1 Amendment No. 213 and Unit No. 2 Amendment No. 163, respectively.

Based on the above evaluation, the NRC staff concludes that the licensee meets Sections 9.1.2 and 9.2.3 of NEI 94-01, Revision 2-A.

3.2.2 Containment Examination Program Results

The licensee stated that 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. Each 10-year interval is divided into three periods of approximately equal duration ensuring that the requisite frequency is maintained.

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

The following tables, reproduced from the licensee's August 26, 2014, application, provide an approximate schedule for the general visual inspection of the accessible containment vessel surfaces, assuming the Type A test frequency is 15 years.

St. Lucie, Unit No. 1

Refueling Outage	Year	ILRT	General Visual Examination of Accessible Internal Surface	General Visual Examination of Accessible External Surface
SLI – 20	F2005	X	X	X
SLI – 21	S2007			
SLI – 22	F2008		X	X
SLI – 23	S2010			
SLI – 24	W2011		X	X
SLI – 25	F2013			
SLI – 26	S2015			
SLI – 27	F2016		X	
SLI – 28	S2018			X
SLI – 29	F2019	X	X	X
SLI – 30	S2021			

St. Lucie, Unit No. 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 licensee also stated that 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 the TSs. 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.

The licensee stated in the LAR that accessible Service Level I 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, this serves to protect the containment vessel from deterioration.

The licensee stated in the LAR that inspection results indicate that no significant corrosion effects have been experienced involving the containment vessel and penetrations. The NRC staff agrees that 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.

3.2.3 <u>Type B and Type C LLRT Program</u>

NUREG-1493, "Performance-Based Containment Leak-Test Program, Final Report," dated September 30, 1995 (Reference 18), indicates that Type B and Type C tests can identify the vast majority of all containment leakage paths. While the licensee's LAR adopts the guidance in NEI 94-01, Revision 2-A, in place of NEI 94-01, Revision 0, the licensee otherwise does not alter the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing at St. Lucie will continue to provide a high degree of assurance that containment leakage rates are maintained well within limits.

The licensee stated that the St. Lucie 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 Part 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 and placed on a test frequency of each outage (or 24 months). To permit operation at EPU conditions, the NRC issued Amendment No. 213 (ADAMS Accession No. ML12156A208), which raised the SL-1 P_a value from 39.6 psig to 42.8 psig, and Amendment No. 163 (ADAMS Accession No. ML12235A463), which raised the SL-2 P_a 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 P_a pressure and to conservatively establish a new performance history, even though the new P_a values are not significantly higher.

In its May 14, 2015, letter, the licensee stated that the most recent ILRTs on SL-1 and 2 met the requirements for Type A test (ILRT) pressure in conjunction with the current values for P_a developed from EPU and bound the revised P_a values of Unit No. 1 Amendment No. 213 and Unit No. 2 Amendment No. 163, respectively. The licensee also provided the following four tables in its May 14, 2015, letter, demonstrating for the most recent ILRTs on SL-1 and 2 that the requirement of American National Standards Institute (ANSI)/American Nuclear Society (ANS)-56.8-1994, Section 3.2.11, Type A test pressure was met: "The Type A test pressure shall not be less than 0.96 P_a nor exceed P_d [design pressure]... The test pressure shall be established relative to the external pressure of the primary containment measured at the start of the Type A test."

Unit No. 1 ILRT Pressure Sequence

Date	Time	Pressure	Description
12/07/2005	08:25	42.79 psig	Secured air compressors
12/08/2005	00:41	41.43 psig	Start Type A test
12/08/2005	08:48	41.29 psig	End Type A test
12/08/2005	13:35	41.18 psig	End verification test

P_a = 42.8 psig, 0.96 P_a = 41.09 psig (from Unit No. 1 Amendment No. 213)

Unit No. 2 ILRT Pressure Sequence

Date	Time	Pressure	Description
12/09/2007	23:30	43.11 psig	Secured air compressors
12/10/2007	13:05	42.09 psig	Start Type A test
12/10/2007	21:05	41.94 psig	End Type A test
12/11/2007	01:15	41.85 psig	End verification test

P_a = 43.48 psig, 0.96 P_a = 41.74 psig (from Unit No. 2 Amendment No. 163)

Unit No. 1 - Cumulative Types B & C Test (LLRT) Totals

Outage	Shutdown Date	P_a (psig)	As-Found Bypass (sccm)	As-Found Total (sccm)	As-Left Bypass (sccm)	As-Left Total (sccm)	Notes
SL1-25	09/2013	42.8	14,567	198,065	22,126	55,144	
SL1-24	11/2011	42.8	20,088	27,599	22,951	56,825	
SL1-23	04/2010	42.8	17,573	47,803	34,344	50,294	
SL1-22	10/2008	39.6	16,688	89,879	31,698	62,718	
SL1-21	04/2007	39.6	26,678	39,546	43,054	60,629	
SL1-20	10/2005	39.6	15,450	28,570	51,584	103,593	ILRT performed

Total (Types B & C) Limit = 544,933 sccm, Bypass Limit = 87,189 sccm

Unit No. 2 - Cumulative Type B & C Test (LLRT) Totals

Outage	Shutdown Date	P_a (psig)	As-Found Bypass (sccm)	As-Found Total (sccm)	As-Left Bypass (sccm)	As-Left Total (sccm)	Notes
SL2-21	03/2014	43.5	29,921	65,838	35,403	112,690	
SL2-20	08/2012	43.5	32,608	72,035	39,710	87,729	
SL2-19	01/2011	41.8	17,768	49,094	29,053	104,099	
SL2-18	04/2009	41.8	17,832	58,714	29,927	85,259	
SL2-17	09/2007	41.8	53,624	83,860	20,883	73,634	ILRT performed
SL2-16	04/2006	41.8	39,833	69,685	48,542	88,855	

Total (Types B & C) Limit = 585,233 sccm, Bypass Limit = 93,637 sccm

Attachment 6 of the application contains a listing of the respective LLRT failures dating back to the period of the last ILRT. Section 3.2 of the application notes that, "When a component fails to meet its respective administrative limit, it is evaluated using the St. Lucie corrective action process and placed on a test frequency of each outage (or 24 months)." After the component is repaired and retested, it is returned to operable status for a mode in which it is required. The NRC staff requested for the "PSL-1 [Plant St. Lucie, Unit No. 1] and PSL-2 LLRT Failures 2005-2014" listed in Attachment 6 that the licensee explain what corrective actions have been taken to correct the problems of components with a history of repetitive failures. In its May 14, 2015, letter, the licensee provided a listing of corrective actions taken for valves that have had historical repetitive failures. The May 14, 2015, letter provided the NRC staff with assurance that the corrective action process with respect to LLRT failures is effectively being managed.

The NRC staff concludes that the results of the past containment leakage testing and the containment examination programs demonstrate acceptable performance of the St. Lucie containment and demonstrate that the structural and leak-tight integrity of the containment structure is adequately managed. The structural and leak-tight integrity of the St. Lucie containment will continue to be periodically monitored and managed by the aforementioned containment leakage testing and containment examination programs.

3.3 Conformance to NEI 94-01, Revision 2-A

As required by 10 CFR 50.54(o), the SL-1 and 2 containments are subject to the requirements set forth in 10 CFR Part 50, Appendix J. Option B of Appendix J requires that test intervals for

Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the SL-1 and 2, Appendix J Testing Program Plans are based on RG 1.163, which endorses NEI 94-01, Revision 0. The licensee's application, as amended, proposes to revise the Appendix J Testing Program Plan by implementing the guidance in NEI 94-01, Revision 2-A.

By letter dated June 25, 2008 (Reference 11), the NRC published a safety evaluation (SE) with limitations and conditions for NEI 94-01, Revision 2. In the SE, the NRC staff concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR Part 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE. In particular, the SE included provisions for extending the ILRT Type A interval to 15 years, subject to the limitations and conditions provided in Section 4.1 of the SE. The NRC staff noted in the SE that NEI 94-01, Revision 2, incorporates the regulatory positions stated in RG 1.163. The accepted version of NEI 94-01, Revision 2, was subsequently published as Revision 2-A of NEI TR 94-01 on November 19, 2008 (Reference 5).

The licensee stated that SL-1 and 2 will meet the limitations and conditions of NEI 94-01, Revision 2-A, Section 4.1. Accordingly, St. Lucie will be adopting the testing criteria of ANSI/ANS 56.8–2002 (Reference 12), rather than the criteria of ANSI/ANS 56.8–1994 (Reference 13).

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

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

Since the application, as amended, will invoke NEI 94-01, Revision 2-A, as the reference for TS 6.8.4.h, the licensee is not seeking to extend the frequencies of the Type C performance-based tests beyond 60 months.

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

3.3.1 <u>Condition 1</u>

Condition 1 states that for calculating the Type A leakage rate, the licensee should use the definition in NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002 (the NRC staff's SE in Reference 10, Section 3.1.1.1).

The licensee stated: "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."

Section 3.2.9, "Type A Test Performance Criterion," of ANSI/ANS-56.8-2002 (Reference 12) defines the "performance leakage rate" and states, in part:

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

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

> Section 5.0 of NEI TR 94-01, Revision 2, uses a definition of "performance leakage rate" for Type A tests that is different from that of ANSI/ANS-56.8-2002 (Reference 11). The definition contained in NEI TR 94-01, Revision 2, is more inclusive because it considers excessive leakage in the performance determination. In defining the minimum pathway leakage rate, NEI TR 94-01, Revision 2, includes the leakage rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position prior to the performance of the Type A test. Additionally, the NEI TR 94-01, Revision 2, definition of performance leakage rate requires consideration of the leakage pathways that were isolated during performance of the test because of excessive leakage in the performance determination. The NRC staff finds this modification of the definition of "performance leakage rate" used for Type A tests to be acceptable.

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

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

The NRC staff reviewed the definitions of "performance leakage rate" contained in NEI 94-01, Revision 2, Revision 2-A, and Revision 3-A (Reference 15). The NRC staff concludes that the definitions contained in all three revisions are identical. Based on this, the NRC staff agrees with the licensee that, "Revision 3-A contains the same definition as Revision 2." Therefore, the NRC staff concludes that both SL-1 and 2 Containment Leakage Rate Testing Programs use the definitions found in Section 5.0 of NEI 94-01, Revision 2, and Revision 2-A, for calculating the Type A leakage rate in the St. Lucie Containment Leakage Rate Testing Program. Accordingly, Condition 1 is satisfied.

3.3.2 Condition 2

Condition 2 states that the licensee submit a schedule of containment inspections to be performed prior to and between Type A tests (the NRC staff's SE in Reference 10, Section 3.1.1.3).

The licensee provided a schedule of containment inspections in Section 3.3 of the application.

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

> NEI TR 94-01, Revision 2, Section 9.2.3.2, states that: "To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years." NEI TR 94-01, Revision 2, recommends that these inspections be performed in conjunction or coordinated with the examinations required by ASME Code, Section XI, Subsections IWE and IWL. The NRC staff finds that these visual examination provisions, which are consistent with the provisions of regulatory position C.3. of RG 1.163, are acceptable considering the longer 15 year interval. Regulatory Position C.3 of RG 1.163 recommends that such examination be performed at least two more times in the period of 10 years. The NRC staff agrees that as the Type A test interval is changed to 15 years, the schedule of visual inspections should also be revised. Section 9.2.3.2 in NEI TR 94-01, Revision 2, addresses the supplemental inspection requirements that are acceptable to the NRC staff.

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

> To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years. It is recommended that these inspections be performed in conjunction or coordinated with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL required examinations.

The NRC staff reviewed the following parts of the application:

- Enclosure Section 3.3, which provides a summary of past and future SL-1 and 2 Containment IWE Supplemental Inspection schedules.
- Attachment 5, Section 1.2, which provides a summary of the scope of the SL-1 and 2 Containment ISI programs.

The NRC staff performed a parallel review of the SL-1 and 2 "Containment Examination Program" and concludes that for Condition 2, the general visual inspections requirements meet the criteria noted in NEI 94-01, Revision 2-A. Based on the observation from this parallel review, the NRC staff concludes that Condition 2 is satisfied.

3.3.3 Condition 3

Condition 3 states that the licensee is to address the areas of the containment structure potentially subjected to degradation (the NRC staff's SE in Reference 10, Section 3.1.3).

The licensee states:

> 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.

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

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

The NRC staff reviewed the following parts of the licensee's application:

- Enclosure Section 3.4, which provides summary statements of identified deficiencies from the SL-1 and 2 historical IWE containment inspections.
- Attachment 5, Section 1.2, which provides a summary of the scope of the SL-1 and 2 Containment ISI programs.

The NRC staff performed a parallel review of the SL-1 and 2 "Containment Examination Program" and concludes that the licensee met the specifications of this condition. Accordingly, Condition 3 is satisfied.

3.3.5 Condition 4

Condition 4 states that the licensee is to addresses any tests and inspections performed following major modifications to the containment structure, as applicable (the NRC staff's SE in Reference 10, Section 3.1.4):

> 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.

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

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

The licensee indicated in Section 3.1 of the application that:

> There are no containment modifications planned for Unit I 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.

Section 3.1 of the licensee's application also documents the SL-1 and 2 ILRT (Type A test) results that followed the restoration of the SL-1 and 2 containment construction hatches during SL1-20 (fall 2005) and SL2-17 (fall 2007), respectively. The December 2005 periodic Type A test for Unit No. 1 yielded a "... performance leak rate corresponding to the definition in NEI 94-01 was 0.2102% wt/day ..." The December 2007 periodic Type A test for SL-2 yielded a "... performance leak rate corresponding to the definition in NEI 94-01 was 0.1930% wt/day ..."

TS 6.8.4.h imposes the overall limit for maximum allowed containment leakage rate, L_a at P_a, at 0.50 percent of containment air weight per day.

TS 6.8.4.h.a states:

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

Therefore, the maximum limit for SL-1 or 2 startup following completion of Type A testing is ≤ 0.75 L_a, which equals 0.375 percent of containment air weight per day.

Since the performance leakage rates documented in both the SL-1 December 2005 Type A test and the SL-2 December 2007 Type A test were less than 0.375 percent of containment air weight per day, the NRC staff concludes that there is reasonable assurance that Condition 4 will be satisfied in the future based on past performance of the units.

3.3.5 Condition 5

Condition 5 specifies that the normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2-A, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition (the NRC staff's SE in Reference 10, Section 3.1.1.2).

In the application, the licensee stated that it acknowledges and accepts the NRC staff position as communicated to the nuclear industry in Regulatory Issue Summary (RIS) 2008-27 dated December 8, 2008.

Section 3.1.1.2 of the NRC staff SE in Reference 10, "Deferral of Tests Beyond The 15-Year Interval," states, in part:

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

The NRC staff notes that the licensee has acknowledged and accepted the NRC staff position discussed in Condition 5. The NRC staff finds that the licensee has confirmed, by referencing RIS 2008-27 (Reference 14) in its statement of compliance, its understanding that any extension of the Type A test interval beyond the upper-bound performance-based limit of 15 years should be infrequent and should be requested only for compelling reasons, and that the NRC staff will implement the position in RIS 2008-27 in reviewing such requests. Accordingly, the NRC staff concludes that Condition 5 is satisfied.

3.3.6 Condition 6

Condition 6 states that 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 Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, including the use of past containment ILRT data.

Condition 6 is not applicable to SL-1 and 2 because neither unit is licensed pursuant to 10 CFR Part 52. SL-1 and 2 were licensed under 10 CFR Part 50.

3.4 Probabilistic Risk Assessment of the Proposed Change

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 3-A (Reference 15), states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01 states that the assessment should be performed using the approach and methodology described in EPRI Technical Report 1018243[1], "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

In its safety evaluation report (SER) dated June 25, 2008 (Reference 11), the NRC staff found the methodology in NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. These conditions, set forth in Section 4.2 of the SER for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submits documentation indicating that the technical adequacy of its Probabilistic Risk Assessment (PRA) is consistent with the guidance of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," relevant to the ILRT extension application. Additional application specific guidance on the technical adequacy of a PRA used to extend ILRT intervals is provided in the SER for EPRI TR-1009325, Revision 2.

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

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[1] EPRI TR-1018243, is also identified as EPRI TR-1009325, Revision 2-A. This report is publicly available and can be found at www.epri.com by typing "1018243" in the search field box.
[2] Section 4.2 of the SER for EPRI TR-1009325, Revision 2, indicates that the clarification regarding small increases

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

4. An LAR is required in instances where containment overpressure is relied upon for emergency core cooling system (ECCS) performance.

3.4.1 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval to once in 15 years. The risk assessment was provided in Attachment 5 of the application. Additional information was provided by the licensee in its letter dated February 6, 2015.

In Section 3.5.1 of the application, the licensee stated that the plant-specific risk assessment follows the guidance in NEI 94-01, Revision 3-A[3]; the methodology described in EPRI TR-1018243, Revision 2-A of 1009325; and the NRC regulatory guidance outlined in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Additionally, the licensee used the methodology from Calvert Cliffs Nuclear Power Plant to assess the risk from undetected containment leaks due to steel liner corrosion.

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

3.4.2 Technical Adequacy of the PRA

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

Consistent with the information provided in Regulatory Issue Summary (RIS) 2007-06 (ADAMS Accession No. ML070650428), "Regulatory Guide 1.200 Implementation," the NRC staff will use Revision 2 of RG 1.200 (ADAMS Accession No. ML090410014) to assess technical adequacy of the PRA used to support risk-informed applications received after March 2010. In Section 3.2.4.1 of the SER for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, the NRC staff states that Capability Category I of the ASME PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution

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in risk is provided in Section 3.2.4.5; however, the clarification is actually provided in Section 3.2.4.6.

[3] NEI 94-01, Revision 3-A (ADAMS Accession No. ML12221A202), added guidance for extending Type C LLRT surveillance intervals beyond 60 months. The guidance for extending Type A ILRT surveillance intervals beyond 10 years is the same as that in Revision 2-A.

among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

Section 3.5.2, "PRA Quality," of the enclosure to the application states that the St. Lucie internal events PRA is a Level 1 and Level 2 model. The licensee states that the model is maintained and upgraded in accordance with the plant 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.

Section 3.5.2 of the enclosure to the application also states that, "the most recent PRA self-assessment ensured compliance with RG 1.200, Revision 2 by identifying gaps in relation to the [ASME/ANS] PRA standard as endorsed by the Regulatory Guide." The gap assessment was submitted as a supplement (Reference 19) to the licensee's application to adopt the National Fire Protection Association (NFPA) Standard 805 (NFPA 805). The licensee states that the gap assessment revealed that the current open findings and observations (F&Os) are related to Interfacing System Loss of Coolant Accident (ISLOCA) analysis and respective models. The licensee concludes that changes to ISLOCA contributions have minimal impact on the risk analysis for the ILRT frequency extension application because the ISLOCA models are conservative. The NRC staff agrees that the open ISLOCA F&Os have minimal impact on the current application, because their resolution is expected to result in lower frequency for EPRI Accident Class 8 related to containment bypass and the risk metrics associated with the ILRT extension application, change in LERF, population dose and conditional containment failure probability (CCFP), are only minimally affected by the lower Class 8 frequency.

In Section 3.2.4.2 of the SER for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, the NRC staff states that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

The licensee performed an analysis of the impact of external events in Section 6.3 of Attachment 5 to the application. The licensee's assessment addressed fire, seismic, high winds, external flooding, and nearby facility and transportation accidents.

The licensee based the ILRT plant-specific risk assessment on the NFPA 805 fire risk values, quoting fire CDF values of 5.12E-05/year for SL-1 and 6.70E-05/year for SL-2, and fire LERF values of 7.00E-06/year for SL-1 and 7.90E-06/year for SL-2. The NRC staff questioned the use of NFPA 805 fire risk values for the ILRT extension application because the licensee's NFPA 805 application is currently under review by the NRC staff, and these fire risk values reflect the state of the plant after the transition to NFPA 805. The licensee also provided previous estimates of fire risk performed in 2010 in support of the application for EPU amendment (ADAMS Accession Nos. ML013530273 and ML013600080), indicating that a realistic assessment of CDF is less than 5.0E-06/year and LERF less than 5.0E-07/year. The NRC staff finds these previous fire risk estimates acceptable for the ILRT extension application. Because the NFPA 805 fire risk values used in the ILRT extension application bound these previous estimates included in the EPU application, the NRC staff finds the use of NFPA 805 fire risks reasonable for the ILRT risk assessment.

The licensee states that SL-1 and 2 are located in a zone of low seismicity and that the seismic assessment from Individual Plant Examinations for External Events (IPEEE) implemented only a limited scope screening that did not provide a CDF value. In order to get an order-of-magnitude estimate of seismic CDF required for the ILRT frequency extension application, the licensee performed a simplified site-specific assessment. The licensee used the methods and values for plant fragility curves consistent with the NRC study published in "Results of Safety/Risk Assessment of Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants'" (ADAMS Accession No. ML100270582). The licensee states that it used the updated seismic hazard estimates generated by EPRI for St. Lucie within the framework of the Post-Fukushima Recommendation 2.1 based on 2012 Central and Eastern United States and the 2013 EPRI Ground Motion Model (provided by the licensee's March 31, 2014 letter). Further, the licensee estimated a seismic LERF by assuming that early large releases will occur at a rate of 10 percent of the CDF. The NRC staff finds the licensee's estimate of seismic CDF reasonable for the application, because the St. Lucie site is considered to have low seismicity and the methods used by the licensee are consistent with the methods used in the NRC GI-199 study referenced above. The NRC staff finds the licensee's estimate of seismic LERF reasonable for the ILRT extension application.

The licensee also provided an estimate of risk related to high winds, external floods, and transportation events and other hazards. The licensee states that during IPEEE, all of these hazards were screened, and no core damage or LERF frequencies were developed. The licensee provided an upper bound estimate of CDF and LERF, which was performed to support the EPU amendment request. The licensee's estimate relies on the assumption that these external events are likely to result in a loss of offsite power (LOOP) initiating event, where power recovery in the short-term is unlikely. The licensee used the frequency for weather-induced LOOP as bounding initiating event and provided estimates of CDF and LERF. The NRC staff finds the licensee's assessment of high winds, external floods, transportation events, and other hazards reasonable for the ILRT extension application.

The licensee evaluated its PRA against the current ASME/ANS PRA Standard and Revision 2 of RG 1.200, evaluated the gaps for applicability to the ILRT interval extension, and explained their impact. The NRC staff reviewed these gaps and agrees that the cumulative impact of all open gaps has no significant impact on the ILRT interval extension application. Furthermore, the licensee included a quantitative assessment of the contribution of external events. Therefore, the NRC staff concludes that the PRA model used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequencies. Accordingly, the first condition is met.

3.4.3 Estimated Risk Increase

The second condition stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small, and consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.5 of the NRC SER for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-(roentgen equivalent man) 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 that the increase in CCFP be less than or equal to 1.5 percentage points. Additionally, for plants that rely on containment over-pressure for net positive suction for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. As discussed in Section 3.4.5 below, SL-1 and 2 do not credit containment overpressure for ECCS performance. Thus, the risk metrics relevant to the ILRT application include LERF, population dose, and CCFP.

The licensee reported the results of the plant-specific risk assessment in Section 3.5.3 of the application. Details of the risk assessment are provided in Attachment 5 of the application. The reported risk impacts are based on a change in test frequency from three tests in 10 years (the test frequency under 10 CFR Part 50 Appendix J, Option A) to one test in 15 years. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A ILRT frequency:

1. The reported increase in LERF for internal events is 4.49E-08/year for SL-1 and 6.28E-08/year for SL-2. These changes in internal events risk are considered to be "very small" (i.e., below 1E-07/year) per the acceptance guidelines in RG 1.174. The reported increase in LERF for combined internal and external events (per Attachment 5, Table 6-4 of the application) is 4.87E-07/year for SL-1 and 6.36E-07/year for SL-2. The risk contribution from external events includes the effects of internal fires, seismic events, high winds, external floods and transportation events and other hazards, as discussed in Section 3.4.2 above. These changes in risk are considered to be "small" (i.e., between 1E-06/year and 1E-07/year) per the acceptance guidelines in RG 1.174. An assessment of total baseline LERF is required to show that the total LERF is less than 1E-05/year. The total LERF including internal and external events prior to the ILRT extension is estimated by the licensee to be 8.32E-06/year for SL-1 and 8.52E-06/year for SL-2 (per LAR Attachment 5, Table 6-3). Therefore the total LERF for internal and external events after the ILRT extension is 8.81E-06/year for SL-1 and 9.16E-06/year for SL-2. The total LERF, given the increase in ILRT interval, is below 1E-05/year for both units.

2. The reported change in Type A ILRT frequency from three in 10 years to once in 15 years results in a reported increase in the total population dose of 2.67E-02

person-rem/year for SL-1 and 3.74E-02 person-rem/year for SL-2. These values are based on the population dose data from NUREG/CR-4551 for Surry, adjusted to reflect St. Lucie power level, allowable containment leak rate and population around the site, as stated in Section 4.2.4 of Attachment 5 of the application. Using site-specific dose estimates the licensee estimated increases in total population dose of 4.76E-02 person-rem/year for SL-1 and 6.65E-02 person-rem/year for SL-2, as indicated in Section 6.4 of Attachment 5 of the application. The licensee states that the site-specific dose estimates are based on analyses performed in support of license renewal assessment of SAMA, adjusted to reflect the current St. Lucie power level and population increase estimates to year 2040. The reported increase in total population dose is below the values provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2. Thus, this increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed ILRT interval extension amendment.

3. The increase in CCFP due to change in test frequency from three in 10 years to once in 15 years is 0.83 percent for SL-1 and 0.90 percent for SL-2. This value is below the acceptance guidelines in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2.

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

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

The third condition stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2, acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensee shall be 100 L_a instead of 35 L_a. As noted by the licensee in the table in Section 3.5.1 of enclosure to the application, the methodology in EPRI TR-1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the St. Lucie plant-specific risk assessment. Accordingly, the third condition is met.

3.4.5 Applicability if Containment Overpressure is Credited for ECCS Performance

The fourth condition stipulates that in instances where containment over-pressure is relied upon for ECCS performance, an amendment request is required to be submitted. In Section 3.5.1 of enclosure to the application, the licensee stated that St. Lucie does not credit containment overpressure for net positive suction head of the ECCS pumps. Accordingly, the fourth condition is not applicable.

3.5 Summary of Technical Evaluation

Based on the NRC staff review and evaluation of the licensee's application, as supplemented, the NRC staff finds that there is reasonable assurance that the licensee has addressed the requirements and guidance to demonstrate acceptability of adopting TR NEI 94-01, Revision 2-A. The NRC staff also determines that the structural and leak-tight integrity of the SL-1 and 2 containments will continue to be monitored and maintained after the performance-based Type A test intervals are extended up to 15 years. Therefore, the NRC staff concludes that it is acceptable for SL-1 and 2 to (i) revise TS 6.8.4.h to adopt NEI 94-01, Revision 2-A, as the implementation document, and (ii) extend on a permanent basis the current Type A test interval up to 15 years

4.0 STATE CONSULTATION

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

5.0 PUBLIC COMMENTS

On January 20, 2015, the NRC staff published its regular biweekly notice in the Federal Register regarding "Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations" (80 FR 2747). With respect to amendments proposed to be issued, in accordance with the requirements in 10 CFR 50.91(a)(2)(ii), the notice provided a 30-day period for public comment on the staff's proposed determination that the associated amendments do not involve a significant hazards consideration (NSHC). The notice included the NRC staff's proposed NSHC determinations for several different nuclear power plants, including the proposed amendments for SL-1 and 2.

Comments were received in response to the *Federal Register* notice (ADAMS Accession No. ML15027A337). The comments raised concerns regarding the public availability of the final safety analysis reports for nuclear power plants in general. However, the comments did not cite any of the specific proposed amendments included in the notice. In addition, the issues discussed in the public comments do not specifically pertain to the proposed NSHC determination for any of the specific nuclear power plants included in the *Federal Register* notice. As such, the NRC staff is not providing a response to the comments.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments change inspection or surveillance requirements with respect to use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff determined that the amendments involve no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. By *Federal Register* notice dated January 20, 2015 (80 FR 2750), the Commission previously issued a proposed finding that the amendments involve NSHC, and there has been no public comment on these findings. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. FPL letter (L-2014-230), dated August 26, 2014, "Application for Technical Specifications Change to Permanently Extend the Integrated Leak Rate Test (ILRT) Frequency to 15 Years" (ADAMS) Accession No. ML14241A496).

2. FPL letter (L-2015-015), dated January 14, 2015, "Response to Request for Additional Information (RAI) Regarding License Amendment Request to Permanently Extend the Integrated Leak Rate Test (ILRT) Frequency to 15 Years (MF4694 and MF4695)" (ADAMS Accession No. ML15029A496).

3. FPL letter (L-2015-015), dated February 6, 2015, "Response to Request for Additional Information (RAI) Regarding License Amendment Request to Permanently Extend the Integrated Leak Rate Test (ILRT) Frequency to 15 Years (MF4694 and MF4695)" (ADAMS Accession No. ML15118A541).

4. FPL letter (L-2015-149), dated May 14, 2015, "Response to Request for Additional Information (RAI) Regarding License Amendment Request to Permanently Extend the Integrated Leak Rate Test (ILRT) Frequency to 15 Years (MF4694 and MF4695)" (ADAMS Accession No. ML15140A081).

5. NEI 94-01, Revision 2-A, dated November 19, 2008, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML100620847).

6. FPL letter (L-2015-143), dated August 15, 2002, "Risk-Informed One Time Increase in Integrated Leak Rate Test Surveillance Interval" (ADAMS Accession No. ML022330608).

7. NRC letter to FPL, dated April 10, 2003, "St. Lucie Units 1 and 2 - Issuance of Amendments Regarding Risk-Informed Integrated Leak Rate Testing Extension (TAC Nos. MB6138 and MB6139)" (ADAMS Accession No. ML031000752).

8. NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program" (ADAMS Accession No. ML003740058).

9. NEI TR 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML11327A025).

10. NEI TR NEI 94-01, Revision 2, dated August 31 2007, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" and Electric Power Research Institute Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (ADAMS Accession No. ML072970206).

11. NRC Staff Safety Evaluation, dated June 25, 2008, "Final Safety Evaluation for Nuclear Energy Institute Topical Report (TR) 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" And Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (TAC No. MC9663)" (ADAMS Accession No. ML081140105).

12. ANSI/ANS-56.8-2002, Reaffirmed August 9, 2011, "Containment System Leakage Testing Requirements."

13. ANSI/ANS-56.8-1994, Approved August 4, 1994, "Containment System Leakage Testing Requirements."

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

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

16. NRC "St. Lucie Plant, Unit 1 - Issuance of Amendment Regarding Extended Power Uprate (TAC No. ME5091)," dated July 9, 2012 (ADAMS Accession No. ML12156A208).

17. NRC "St. Lucie Plant, Unit 2 - Issuance of Amendment Regarding Extended Power Uprate (TAC No. ME5843), dated September 24, 2012 (ADAMS Accession No. ML12235A463).

18. NUREG-1493, "Performance-Based Containment Leak-Test Program, Final Report," dated September 30, 1995 (ADAMS Legacy Library Accession No. 9510200161).

19. FPL Letter (L-2014-083) dated March 25, 2014, "90-Day Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants" (ADAMS Accession No. ML14114A458).

20. NRC letter dated August 20, 2013, to NEI that the limitations/conditions for NEI 94-01, Revisions 2A are missing in Revision 3A (ADAMS Accession No. ML13192A394).

Principal Contributors: M. Biro
D. Hoang
D. Nold

Date: August 27, 2015

August 27, 2015

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
NextEra Energy
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING CHANGE TO TECHNICAL SPECIFICATION 6.8.4.h, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (TAC NOS. MF4694 AND MF4695)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment Nos. 226 and 176 to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2, respectively. These amendments consist of changes to the technical specifications (TSs) in response to your application dated August 26, 2014, as supplemented by letters dated January 14, February 6, and May 14, 2015.

Each amendment revises TS 6.8.4.h, "Containment Leakage Rate Testing Program," to allow extension of the 10-year frequency of the Type A test (i.e., integrated leak rate test) to a 15-year frequency. The proposed change is accomplished by replacing the previous wording, which references Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," with wording referencing Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [*Code of Federal Regulations*] Part 50, Appendix J," October 2008.

The NRC staff's related safety evaluation of the amendments is enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA by PTam for/
Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50-389

Enclosures:
1. Amendment No. 226 to DPR-67
2. Amendment No. 176 to NPF-16
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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ADAMS Accession No.: ML15195A655 *By memo ML15159B005 **by memo ML15120A505 ***by memo ML15120A025

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DATE	7/24/15	7/23/15	8/20/15	6/9/15	5/20/15
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DATE	05/20/15	8/18/15	8/26/15	8/27/15	

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