April 10, 2003

Mr. J. A. Stall
Senior Vice President, Nuclear and
Chief Nuclear Officer
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: ST. LUCIE UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING RISK-INFORMED INTEGRATED LEAK RATE TESTING EXTENSION (TAC NOS. MB6138 AND MB6139)

Dear Mr. Stall:

The Commission has issued the enclosed Amendment Nos. 187 and 130 to Facility Operating License Nos. DPR-67 and NPF-16 for the Saint Lucie Plant, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 15, 2002, as supplemented December 13, 2002.

These amendments revise TS Section 6.8.4.h, Containment Leakage Rate Testing Program, to allow a one-time 5-year extension to the current 10-year test interval for the containment integrated leak rate test (ILRT). The proposed changes were submitted on a risk-informed basis as described in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The risk-informed analysis supporting the proposed changes indicates that the increase in risk from extending the ILRT test interval from 10 to 15 years is insignificant.

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

Sincerely,

/RA/

Brendan T. Moroney, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50-389

Enclosures:
1. Amendment No. 187 to DPR-67
2. Amendment No. 130 to NPF-16
3. Safety Evaluation

cc w/encls: See next page

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Florida Power and Light Company

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FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 187
License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company (the licensee), dated August 15, 2002, as supplemented December 13, 2002, 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.

2. Accordingly, 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 2.C.(2) to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 187, are hereby incorporated in the license. The licensee 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Allen G. Howe, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 10, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 187

TO FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

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

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

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated August 15, 2002, as supplemented December 13, 2002, 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.

2. Accordingly, 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 2.C.2 to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 130, are hereby incorporated in the license. The licensee 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Allen G. Howe, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 10, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 130

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 187 AND 130

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

FLORIDA POWER AND LIGHT COMPANY, ET AL.

SAINT LUCIE PLANT, UNITS 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By letter dated August 15, 2002, as supplemented December 13, 2002, Florida Power and Light Company, et al., (FPL, the licensee) requested amendments to Operating Licenses DPR-67 and NPF-16 for Saint Lucie (STL) Units 1 and 2, respectively. The amendments would revise Technical Specification (TS) Section 6.8.4.h, Containment Leakage Rate Testing Program, to allow a one-time 5-year extension to the current 10-year test interval for the containment integrated leak rate test (ILRT). Specifically, the proposed TSs state that the first Unit 1 Type A test performed after the May 1993 test shall be no later than May 2008, and the first Unit 2 Type A test performed after the June 1992 test shall be no later than June 2007.

The proposed changes are submitted on a risk-informed basis as described in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The risk-informed analysis supporting the proposed changes indicates that the increase in risk from extending the ILRT test interval from 10 to 15 years is insignificant.

The licensee's supplement dated December 13, 2002, did not affect the original proposed no significant hazards determination, nor expand the scope of the request as noticed in the *Federal Register* on September 17, 2002 (67 FR 58647).

2.0 REGULATORY EVALUATION

RG 1.174 provides guidance on the use of Probabilistic Risk Assessment findings and risk insights in support of licensee requests for changes to a plant's licensing basis, as in requests for license amendments and technical specification changes.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. STL Units 1 and 2, 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, "Performance-Based Containment Leak-Test Program," dated September 1995, with one listed exception. RG 1.163, Section C, "Regulatory Position" states, "licensees intending to comply with the Option B in the amendment to Appendix J should establish test intervals based upon the criteria in Section 11.0 of Nuclear Energy Institute (NEI) 94-01, rather than using test intervals specified in American Nuclear Standards Institute/American Nuclear Society (ANSI/ANS)-56.8-1994." The NEI 94-01, Section 11 states that Type A testing shall be performed at a frequency of at least once every 10 years.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 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 in certain circumstances. The most recent two Type A tests at each STL unit have been successful, so the current interval requirement is 10 years.

3.0 TECHNICAL EVALUATION

3.1 TSs Administrative Controls Section 6.8.4.h, "Containment Leakage Rate Testing Program"

The licensee proposes to revise the STL Units 1 and 2 TSs Administrative Controls Section 6.8.4.h by adding an additional exception.

The revised STL Unit 1, TSs Administrative Control Section 6.8.4.h is as follows (proposed changes are underlined):

> .... 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<u>(s)</u>:
>
> (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.
>
> <u>(b) The first Type A test performed after the May 1993 Type A test shall be performed no later than May 2008.</u>

The revised STL Unit 2, TSs Administrative Control Section 6.8.4.h is as follows (proposed changes are underlined):

> .... 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<u>(s)</u>:
>
> (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.
>
> <u>(b) The first Type A test performed after the June 1992 Type A test shall be performed no later than June 2007.</u>

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-10 specifies an initial test interval of 48 months, but allows an extended interval of 10 years based upon two consecutive successful tests. The NEI guidelines also contain a provision for extending the test interval an additional 15 months under certain circumstances. The leak rate testing requirements of Option B of Appendix J, and the containment inservice inspection (ISI) requirements mandated by 10 CFR 50.55a complement each other in ensuring the leak-tightness and structural integrity of the containment. Therefore, a detailed evaluation related to the ISI of the containment and potential areas of weaknesses in the containment is performed in the following section.

3.2 ISI for Primary Containment Integrity

The STL Units 1 and 2 are Combustion Engineering pressurized-water reactors (PWRs) with a large, dry (ambient) steel primary containment structure. The containment pressure boundary consists of the steel containment shell structure with concrete enclosure building, containment access penetrations, and process piping and electrical penetrations. The integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests as required by 10 CFR Part 50, Appendix J, and the overall leak-tight integrity of the primary containment is verified through an ILRT. These tests are performed to verify the essentially leak-tight characteristics of the containment structure at the design basis accident pressure. The licensee states that there have been six ILRTs performed on STL Unit 1 and four ILRTs performed on Unit 2, all of which have been successful. Based on these successful Type A tests at STL Units 1 and 2, and the requirements of 10 CFR Part 50, Appendix J, Option B, the current interval requirement is 10 years. With the requested extension of the ILRT time interval, the next overall verification of the containment leak-tight integrity will be performed in May 2008 for STL Unit 1 and June 2007 for STL Unit 2. The licensee, in its August 15, 2002, letter, provided information related to the ISI of the containment and potential areas of weaknesses in the containment that may not be apparent in the risk assessment. The licensee also provided information to explicitly address five questions raised by the NRC staff during its review of other similar TS change requests submitted by other licensees. The staff's evaluation of the licensee's response to these questions and requests for additional information is discussed in the following paragraphs.

ISI Program at STL Units 1 and 2: The licensee states that the STL containment vessels are examined in accordance with the requirements of American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWE, 1992 Edition with 1992 Addenda, the plant protective coatings program, and TSs. These inspection processes are described as follows:

The containment ISI program at STL Units 1 and 2 is described in detail in ISI/IWE-PSL-1/2-PROGRAM, 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 ISI/IWE-PSL-1-PLAN, ASME Section XI, Subsection IWE Containment Building Metal Containment Inservice Inspection Plan for STL Unit 1, and ISI/IWE-PSL-2-PLAN, ASME Section XI, Subsection IWE Containment Inservice Plan for STL Unit 2. The program requirements include inspection of containment surfaces, pressure retaining welds, bolting, and components, seals, gaskets, and moisture barriers using visual, surface, and volumetric techniques as required. Examinations that detect flaws or evidence of degradation are documented through the site corrective action program and are dispositioned in accordance with the requirements of IWE-3000. Personnel performing nondestructive examination (NDE) are qualified and certified in accordance with IWA-2300 of the 1992 Edition with 1992 Addenda of the ASME Code, Section XI and implemented by the procedure CSI-QI-9.1, Qualification and Certification of Nondestructive Examination Personnel. The program complies with the requirements of 10 CFR 50.55a.

The IWE program performs inspection of the entire accessible interior surface of the containment in each of three, 40-month periods within a 10-year surveillance interval. The 100 percent general surface area inspection for the first period on Unit 1 was completed in April 2001. The 100 percent general surface area inspections were completed for the first period in April 2000 and second period in November 2001 on Unit 2. One-third of the moisture barriers at the concrete floor vessel interface on both sides of the containment are inspected during each period. To date, two-thirds of the Unit 1 moisture barrier and one-third of the Unit 2 moisture barrier have been inspected. Unit 2 will be two-thirds complete following the upcoming spring 2003 refueling outage.

Inspection results indicate that no significant corrosion effects have been experienced on the containment vessels. 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 licensee states that during ISI, there has been no indication of containment vessel metal degradation on either unit resulting from the ISI.

The Protective Coating Program at STL requires that a walkdown of the containment interior be performed each refueling outage by the FPL coating specialist and engineering personnel to inspect any existing areas of nonqualified coatings and to determine any other areas in need of repair. Personnel familiar with the American Society of Testing and Maintenance coatings standards inspect the accessible exterior containment surface in accordance with plant procedures. Portions of the upper exterior containment vessel surface are not accessible for inspection due to the unavailability of sufficient installed ladders or platforms, so the containment external surface above the floor is not inspected each outage. Inspections of the upper exterior surfaces of both containments have been performed during previous outages. Inspection of the upper section of the exterior side of the containment vessel identified no degraded areas and no potential means by which corrosion would be promoted, such as moisture sources or equipment interface. Those areas identified by inspection that do not meet the acceptance criteria are evaluated and scheduled for repair as necessary. Following repairs, containment vessel coatings are re-examined by certified NDE examiners and the as-left condition is documented. This allows identification of any potential for containment vessel degradation. There have been no indications of significant degradation of the containment vessel base metal.

The licensee states that general visual inspections of both sides of the accessible containment vessel surface and the shield building are performed as required by the TSs in accordance with Quality Instruction QI 10-PR/PSL-5, TSs Surveillance Inspection of Reactor Building. Results of these inspections have not revealed any additional conditions to those already noted other than minor concrete spalling of the shield building.

Implementing IWE-1240 at STL Units 1 and 2: The ASME Code, Section XI, Subsection IWE inspection plan was implemented for STL Unit 1 on April 7, 2000, and for Unit 2 on August 9, 2000. All inspections have been completed for the first period of the 10-year surveillance interval on both STL Units. There are currently no identified areas at either STL Unit 1 or Unit 2 that require augmented inspections in accordance with IWE-1240.

There have been conditions identified by other inspection processes that relate to the material condition of the containment boundary. The first condition involved a problem with cracking of the moisture barrier at the interface of the concrete floor and containment vessel. This was initially documented and evaluated in STL Condition Report (CR) 97-0890. Subsequent inspections have been performed as part of the corrective action process on both units. Material was removed and the containment vessel wall was inspected in areas where the deterioration of the moisture barrier existed. These inspections determined that only light surface corrosion or discoloring existed with pitting noted in some locations. Using the results of several inspections, FPL's assessment has determined that this issue does not affect the structural or leak-tight integrity of the containment vessel. The second condition related to the containment vessel material condition involved external corrosion, due to moisture accumulation from condensate, on the component cooling water penetrations to containment. This was initially documented in CR 97-1799. Corrective actions, inspections, and evaluation of the most affected penetrations have provided objective evidence that the piping degradation is minor and a large thickness margin is available before encroaching upon design requirements. The site corrective action program has been utilized to track additional inspections and long-term corrective action activities. Based on the inspections, repairs, and evaluation of these issues, the licensee determined that augmented inspection was not required in accordance with IWE-1240.

<u>IWE Table-2500-1, Examination Categories E-D and E-G for seals and gaskets, and examination and testing of bolts</u>: The licensee states that in authorized Relief Request IWE-01, seals and gaskets for containment penetrations are tested in accordance with 10 CFR Part 50, Appendix J. Type B tests are required to be performed at a frequency not to exceed 60 months (air locks not to exceed 30 months), in accordance with plant procedures. The extension of the Type A testing does not affect this frequency. Thus, all penetrations utilizing gaskets and seals as part of the primary containment boundary are tested for leak-tight integrity within each 10-year inspection interval.

The licensee states that in another authorized relief request (IWE-02) for torque or tension testing of all bolting not disassembled during the inspection interval, 10 CFR Part 50, Appendix J, Type B testing verifies that the bolt torque or tension remains adequate to ensure the leak-tight integrity of the containment. The extension of the Type A testing does not affect this frequency of the Type B testing which, as previously stated, is required to be performed within each 10-year interval. In addition, it is noted that the exposed surfaces of bolted connections shall be visually examined in accordance with the requirements of ASME Code, Section XI Table IWE-2500-1, Examination Category E-G. A general visual inspection of the entire containment, once each inspection interval, shall be conducted in accordance with 10 CFR 50.55a(b)(2)(ix)(E).

<u>Integrity of stainless steel bellows</u>: In the past, the integrity of two-ply stainless steel bellows at other plants has been found susceptible to trans-granular stress corrosion cracking, and the leakage through them is not detectable by Type B testing (see NRC Information Notice 92-20, Inadequate Local Leak Rate Testing). The licensee states that STL Units 1 and 2 each have five penetration assemblies that incorporate two-ply mechanical bellows. These are the two main feedwater, two main steam, and fuel transfer penetrations. The licensee states that

review of site operating experience reports and plant drawings found that wire mesh is installed between the two-plies of the bellows ensuring that an adequate gap exists to measure leakage when performing the required Type B tests. These bellows have been tested each outage since startup for both units with satisfactory results.

Inspection of embedded side of the containment steel shell: Inspection of some reinforced concrete and steel containment structures have found degradation on the uninspectable (embedded) side of the containment steel shell of the primary containment.
The licensee states that approximately 80 percent of the steel containment vessel is exposed to permit visual inspection. The remaining 20 percent of the containment vessel that is inaccessible for visual inspection includes the area beneath the concrete floor and a small area around the fuel transfer tube. The relative surface areas were approximated using STL Plant drawings, STL Plant Updated Final Safety Analysis Reports and Standard Mathematical Tables. This approximation applies to both STL Units 1 and 2.

The STL Unit 1 and 2 metal containment vessel is surrounded by a concrete shield building with an annular space between them, which permits a general visual inspection of the containment exterior surface. The containment metal is approximately 2-inches thick and the outer surface, which is coated, is in a dry, protected air space. In addition, under normal operating conditions, the containment exterior surface is inherently warmer than its surrounding environment preventing condensation on the surface and, thus, minimizing the potential for a viable corrosion mechanism to develop.

Based on the information provided in the TS change request and the licensee's response to the staff's five general questions included in the request, the staff finds that: (1) the structural integrity of the containment vessel is verified through the periodic ISIs conducted as required by Subsections IWE and IWL of the ASME Code, Section XI, (2) the integrity of the penetrations, containment isolation valves and mechanical bellows are periodically verified through Type B and C tests as required by 10 CFR Part 50, Appendix J and STL Units 1 and 2 TSs, and (3) the potential for large leakage from the areas that cannot be examined by the ISI has been explicitly modeled in performing the risk assessment. In addition, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities in accordance with Article IWE-5000 of the ASME Code, Section XI. Serious degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 and 10 CFR 50.73. From the findings above, the staff concludes that the licensee's ISI program will provide adequate assurance that the containment structural integrity will be maintained during the extended ILRT period.

3.3 Risk Impact of Extending Type A Test Interval

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The assessment was provided to the staff in the August 15, 2002, application for license amendment. Additional analysis and information were provided by the licensee in a letter dated December 13, 2002. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995, provided 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 EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3-in-10 years to 1-in-10 years increased the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the PWR and boiling-water reactor representative plants confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3-in-10 years to 1-in-20 years leads to an "imperceptible" increase in risk on the order of 0.2 percent and a fraction of one person-rem per year.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6}/year and increases in large early release frequency (LERF) less than 10^{-7}/year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3-in-10 year frequency. RG 1.174 also 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. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided an analysis which estimated the risk metrics discussed above and provided a methodology which is consistent with previously approved submittals. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

(a) A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 3-in-10 year test frequency to a 1-in-15 year test frequency, the increase in the total integrated plant risk, in person-rem/year, is estimated to be 0.18 percent for Unit 1 and 0.11 percent for Unit 2. This increase is comparable to that estimated in NUREG-1493, in which it was concluded that a reduction in the frequency of tests from 3-in-10 years to 1-in-20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

(b) The increase in LERF resulting from a change in the Type A test frequency from the original 3-in-10 years to 1-in-15 years is estimated to be 9.4×10^{-8}/year for Unit 1 and 7.7×10^{-8}/year for Unit 2. However, there is some likelihood that the undetected flaw in the containment liner estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE/IWL). The most recent visual examinations of the STL containment were performed in fall 2002 and fall 2001 for Units 1 and 2, respectively. The next scheduled IWE/IWL containment visual examinations are spring 2004 and spring 2003 for Units 1 and 2, respectively. Visual examinations are expected to be effective in detecting large flaws in the visible regions of the containment, and would reduce the impact of the extended test interval on LERF. The licensee performed additional risk analysis to consider the impact of hypothetical corrosion in inaccessible areas of the containment shell on the proposed change. The risk analysis considered the likelihood of an age-adjusted flaw that would lead to a breach of the containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type A test. When possible corrosion of the containment surfaces is considered, the increase in LERF resulting from a change in the Type A test frequency from the original 3-in-10 years to 1-in-15 years is estimated to be 1.1×10^{-7}/year for Unit 1 and 8.3×10^{-8}/year for Unit 2. Therefore, the staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent the acceptance guidelines of RG 1.174.

(c) RG 1.174 also encourages the use of risk analysis techniques to ensure that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of 0.3 percentage points for both units for the cumulative change of going from a test frequency of 3-in-10 years to 1-in-15 years. The staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed amendment.

The staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174.

The NRC staff concludes, based on the considerations discussed above, that the licensee has adequate procedures to examine and monitor potential age-related and environmental degradations of the pressure retaining components of the STL Units 1 and 2 primary containment. The staff also finds that the increase in risk is within the acceptance guidelines of RG 1.174. Thus, granting a one-time 5-year extension to the current 10-year test interval for the containment ILRT is acceptable. Specifically, this would allow the first Unit 1 Type A test performed after the May 1993 test to be no later than May 2008, and the first Unit 2 Type A test performed after the June 1992 test shall be no later than June 2007.

4.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes an inspection or surveillance requirement. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (67 FR 58647, dated September 17, 2002). Accordingly, these 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 these amendments.

6.0 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES:

1. 10 CFR Part 50, Appendix J

2. ASME *Boiler and Pressure Vessel Code*, Section XI, 1992 Edition including 1992 Addenda

3. USNRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995

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

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

6. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

7. Letter from D. E. Jernigan, FPL to NRC, “Proposed License Amendments - Risk-Informed One Time Increase in Integrated Leakage Rate Test Surveillance Interval, St. Lucie Units 1 and 2,” dated August 15, 2002

8. Letter from D. E. Jernigan, FPL to NRC, “Request for Additional Information Response on Proposed License Amendments - Risk-Informed One Time Increase in Integrated Leakage Rate Test Surveillance Interval, St. Lucie Units 1 and 2,” dated December 13, 2002

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