March 30, 1994

Docket Nos. 50-335
and 50-389

DISTRIBUTION
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Mr. J. H. Goldberg
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Goldberg:

SUBJECT: ST. LUCIE UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CONTAINMENT SURVEILLANCE REQUIREMENTS (TAC NOS. M88317 AND M88318)

The Commission has issued the enclosed Amendment Nos. 127 and 66 to Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated October 26, 1993.

These amendments change Technical Specification 4.6.1.2.a, Containment Leakage Surveillance Requirements, to be consistent with the guidance of NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants."

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,

(Original Signed By)
Jan A. Norris, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 127 to DPR-67
2. Amendment No. 66 to NPF-16
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PDII-2	PM:PDII-2	D:PDII-2	OGC	
NAME	ETana	JNorris	HBerkow	EHoller	
DATE	03/11/94	03/14/94	03/14/94	03/17/94	

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Mr. J. H. Goldberg
Florida Power and Light Company

St. Lucie Plant

cc:
Jack Shreve, Public Counsel
Office of the Public Counsel
c/o The Florida Legislature
111 West Madison Avenue, Room 812
Tallahassee, Florida 32399-1400

Senior Resident Inspector
St. Lucie Plant
U.S. Nuclear Regulatory Commission
7585 S. Hwy A1A
Jensen Beach, Florida 34957

Mr. Joe Myers, Director
Div. of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

Harold F. Reis, Esq.
Newman & Holtzinger
1615 L Street, N.W.
Washington, DC 20036

John T. Butler, Esq.
Steel, Hector and Davis
4000 Southeast Financial Center
Miami, Florida 33131-2398

Mr. Thomas R.L. Kindred
County Administrator
St. Lucie County
2300 Virginia Avenue
Fort Pierce, Florida 34982

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
ABB Combustion Engineering, Nuclear Power
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Mr. Bill Passetti
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

Regional Administrator, RII
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, Georgia 30323

Mr. H. N. Paduano, Manager
Licensing & Special Projects
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

D. A. Sager, Vice President
St. Lucie Nuclear Plant
P.O. Box 128
Ft. Pierce, Florida 34954-0128

C. L. Burton
Plant General Manager
St. Lucie Nuclear Plant
P.O. Box 128
Ft. Pierce, Florida 34954-0128

DATED: March 30, 1994

AMENDMENT NO. 127 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1
AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. NPF-16 - ST. LUCIE, UNIT 2

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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 FACILITY OPERATING LICENSE

Amendment No. 127
License No. DPR-67

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 October 26, 1993, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 127, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 30, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 127

TO FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
B 3/4 6-1	B 3/4 6-1

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT VESSEL

CONTAINMENT VESSEL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without CONTAINMENT VESSEL INTEGRITY, restore CONTAINMENT VESSEL INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

   1. All containment vessel penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.1, and

   2. All containment vessel equipment hatches are closed and sealed.

b. By verifying that each containment vessel air lock is OPERABLE per Specification 3.6.1.3.

---

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$, 0.50 percent by weight of the containment air per 24 hours at P_a, (39.6 psig), or

2. $\leq L_t$, 0.32 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, (19.8 psig).

b. A combined leakage rate of $\leq$ 0.60 L_a for all penetrations and valves subject to Type B and C tests as identified in Table 3.6-1 when pressurized to P_a.

c. A combined leakage rate of $\leq$ 0.27 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.27 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50:

a. Type A test shall be performed in accordance with 10 CFR 50 Appendix J, as modified by approved exemptions.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a or 0.25 L_t,

2. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a (39.6 psig) or P_t (19.8 psig).

d. Type B and C tests shall be conducted with gas at P_a (39.6 psig) at intervals no greater than 24 months except for tests involving air locks.

e. The combined bypass leakage rate shall be determined to be < 0.27 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (39.6 psig) during each Type A test.

f. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

SURVEILLANCE REQUIREMENTS (Continued)

g. All Type A test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

3/4.6.1 CONTAINMENT VESSEL

3/4.6.1.1 CONTAINMENT VESSEL INTEGRITY

CONTAINMENT VESSEL INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a (39.6 psig). As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq$ 0.75 L_a or $\leq$ 0.75 L_t (as applicable) during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50 as modified by approved exemptions with the option of using any NRC-approved method for performing the leak rate testing and calculating the results.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structural is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.70 psi and 2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break accident conditions.

The maximum peak pressure obtained from a steam line break accident is 41.6 psig. The limit of 2.4 psig for initial positive containment pressure will limit the total pressure to 44.0 psig which is the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment air temperature ensures that the containment vessel temperature does not exceed the design temperature of 264°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 41.6 psig in the event of a steam line break accident. A visual inspection in conjunction with Type A leakage test is sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capbility will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

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

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. 66
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 October 26, 1993, 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. 66 , 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: March 30, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 66

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
B 3/4 6-1	B 3/4 6-1

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1*, 2*, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a, 41.8 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

*In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 41.8 psig, or

2. Less than or equal to L_t, 0.35 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 20.9 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

c. A combined bypass leakage rate of less than or equal to 0.12 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.12 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than or equal to 0.60 L_a, and the bypass leakage rate to less than or equal to 0.12 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50:

a. Type A test shall be performed in accordance with 10 CFR 50 Appendix J, as modified by approved exemptions.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a or 0.25 L_t.

   2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage rate at P_a, 41.8 psig or P_t, 20.9 psig.

d. Type B and C tests shall be conducted with gas at P_a, 41.8 psig at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.7.3 and 4.6.1.7.4.

SURVEILLANCE REQUIREMENTS (Continued)

f. The combined bypass leakage rate shall be determined to be less than or equal to 0.12 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 41.8 psig during each Type A test.

g. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

h. The provisions of Specification 4.0.2 are not applicable.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50 as modified by approved exemptions with the option of using any NRC-approved method for performing the leak rate testing and calculating the results.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.7 psi and (2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 43.4 psig. The limit of 0.4 psig for initial positive containment pressure will limit the total pressure to 43.99 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment temperature does not exceed the design temperature of 264°F during steam line break conditions and is consistent with the safety analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 44.0 psig in the event of a steam line break accident. A visual inspection in conjunction with Type A leakage test is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 48-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes devices to lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 48-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 127 AND 66

TO FACILITY OPERATING LICENSE NO. DPR-67 AND NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By letter dated October 26, 1993, Florida Power & Light Company (FPL or the licensee) proposed license amendments to change the Technical Specifications (TS) for the St. Lucie Plant, Units 1 and 2, (St. Lucie or the facility). The proposed amendments would revise the surveillance test schedule in TS 4.6.1.2.a and the associated Bases for performing Type A tests which determine the overall integrated containment leakage rate. The licensee proposed these amendments to provide for operational flexibility by matching the test schedule with the longer fuel cycle lengths and longer refueling outages in which major plant modifications are generally implemented. The proposed TS wording would be consistent with the standard TS for Combustion Engineering Plants (NUREG-1432).

2.0 EVALUATION

10 CFR 50 Appendix J, Section III, paragraph D states "...three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections."

Consistent with the test interval (three tests within 10 years) set forth in 10 CFR 50 Appendix J, the existing TS 4.6.2.1.a states:

> Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either Pa (39.6 psig) or at Pt (19.8 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

The specific test schedule, at 40 ± 10 month intervals, in the TS does not accommodate longer fuel cycle lengths and longer refueling outages in which major plant modifications are implemented. To avoid unnecessary future TS change requests when the fuel cycle does not match the 40 ± 10 month TS required interval, the licensee proposed to revise TS 4.6.1.2.a to read:

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Type A test shall be performed in accordance with 10 CFR 50 Appendix J, as modified by approved exemptions.

TS Bases 3/4.6 will also be revised to reflect the proposed TS changes. The staff has evaluated the proposed TS changes which are summarized below.

NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants," includes guidance for the performance of Type A tests in accordance with 10 CFR 50 Appendix J. The proposed TS wording is consistent with NUREG-1432. The licensee will continue to perform three Type A tests, at approximately three equal intervals, as required by 10 CFR Appendix J and any change to the leak tests will be accomplished only by means of an NRC-approved exemption request to 10 CFR 50 Appendix J. As a result, the staff finds the proposed TS changes acceptable.

3.0 STATE CONSULTATION

Based upon the written notice of the proposed amendments, the Florida State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 (58 FR 67845). 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.

5.0 CONCLUSION

Based on the staff evaluation in Section 2.0 above, the staff concludes that the proposed Technical Specifications changes are acceptable.

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.

Principal Contributor: J. Norris

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