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

December 28, 2010

Mr. Mano Nazar
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 PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING DELETION OF THE STRUCTURAL INTEGRITY TECHNICAL SPECIFICATION, UPDATE OF ACCIDENT MONITORING INSTRUMENTATION REQUIREMENTS, AND VARIOUS ADMINISTRATIVE CHANGES (TAC NOS. ME3489 AND ME3490)

Dear Mr. Nazar:

The Commission has issued the enclosed Amendment Nos. 210 and 159 to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Units 1 (SL1) and 2 (SL2). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 14, 2009, as supplemented by letter dated July 30, 2010.

These amendments would modify TS requirements to delete the Structural Integrity TS, update Accident Monitoring Instrumentation requirements and make various administrative TS changes. Specifically, these amendments would revise TSs by the following:

1. Deletion of Structural Integrity TS 3/4.4.10 (SL1) and TS 3/4.4.11 (SL2);

2. Relocate the SL2 reactor coolant pump flywheel inspection surveillance requirement 4.4.11 to a new administrative control program 6.8.4.o;

3. Delete TS 6.4.1, "Training" to be consistent with NUREG-1432, Revision 3.0, "Standard Technical Specifications Combustion Engineering Plants" (SL1 and SL2) and relocate the definition of licensed operators from TS 6.3.1 to new TS 6.3.2;

4. Revise Accident Monitoring Instrumentation actions 1, 2, 6, and 7 in SL1 TS 3.3.3.8 and Accident Monitoring Instrumentation actions a and b in SL2 TS 3.3.3.6;

5. Change the action in SL2 TS 3.1.2.6 to be logically correct;

6. Correct a typographical error in SL2 SR 4.3.3.2;

7. Correct section heading on index page VI of SL1 TS;

8. Delete reference to Table 3.6-1 on index page XXIV of SL2 TS; and

9. Delete TS 6.8.4.I.2 and 6.9.1.13 steam generator inspection requirements SL2 TS.

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,

Tracy J. Orf

Tracy J. Orf, 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. 210 to DPR-67
2. Amendment No. 159 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. 210
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 (the licensee), dated December 14, 2009, as supplemented by letter dated July 30, 2010, 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, 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. 210, 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

Brenda Mozafari for

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: December 28, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 210

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace Page 3 of Renewed Operating License DPR-67 with the attached Page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove Pages	Insert Pages
V	V
VI	VI
XIV	XIV
3/4 3-43	3/4 3-43
3/4 4-26	3/4 4-26
6-6	6-6

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 2700 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 210 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.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

TABLE 3.3-11 (continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the Total No. of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 2 - With position indication inoperable, restore the inoperable indicator to OPERABLE status or close the associated PORV block valve and remove power from its operator within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information once per shift to determine valve position.

ACTION 4 - With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to the specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 5 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

ACTION 6 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 7 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

DELETED

6.0 ADMINISTRATIVE CONTROLS

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for:

(1) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975,

(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:

- Bachelor's degree in engineering from an accredited institution; or
- Professional Engineer's (PE) license obtained by successful completion of the PE examination; or
- Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or
- Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.4 DELETED

6.5 DELETED

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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 159
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, et al. (the licensee), dated December 14, 2009, as supplemented by letter dated July 30, 2010, 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, 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. 159, 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

Brenda Mozafari for

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: December 28, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 159

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace Page 3 of Renewed Operating License DPR-16 with the attached Page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove Pages	Insert Pages
VI	VI
XVIII	XVIII
XXIV	XXIV
3/4 1-12	3/4 1-12
3/4 3-24	3/4 3-24
3/4 3-41	3/4 3-41
3/4 4-39	3/4 4-39
6-6	6-6
6-7	6-7
6-15f	6-15f
6-15g	6-15g
6-15h	
6-15i	6-15i
6-15j	6-15j
6-20f	6-20f

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 2700 megawatts (thermal).

Commencing with the startup for Cycle 16 and until the Combustion Engineering Model 3410 Steam Generators are replaced, the maximum reactor core power shall not exceed 89 percent of 2700 megawatts (thermal) if:

a. The Reactor Coolant System Flow Rate is less than 335,000 gpm but greater than or equal to 300,000 gpm, or

b. The Reactor Coolant System Flow Rate is greater than or equal to 300,000 gpm AND the percentage of steam generator tubes plugged is greater than 30 percent (2520 tubes/SG) but less than or equal to 42 percent (3532 tubes/SG).

This restriction in maximum reactor core power is based on analyses provided by FPL in submittals dated October 21, 2005 and February 28, 2006, and approved by the NRC in Amendment No. 145, which limits the percent of steam generator tubes plugged to a maximum of 42 percent (3532 tubes) in either steam generator and limits the plugging asymmetry between steam generators to a maximum of 600 tubes.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 159 are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

SECTION PAGE

LIST OF TABLES (Continued)

REACTIVITY CONTROL SYSTEMS

BORIC ACID MAKEUP PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2 is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 operable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid makeup pump(s) shall be demonstrated OPERABLE by verifying that the pump(s) develop the specified discharge pressure when tested pursuant to the Inservice Testing Program.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

4.3.3.2 At least once per 18 months, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a.* With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

b.* With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

c.** With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

d.** With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status, and

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

e. The provisions of Specification 3.0.4 are not applicable.

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* Action statements do not apply to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

** Action statements apply only to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

DELETED

6.2.3 SHIFT TECHNICAL ADVISOR FUNCTION

An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for:

(1) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975,

(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:

- Bachelor's degree in engineering from an accredited institution; or
- Professional Engineer's (PE) license obtained by successful completion of the PE examination; or
- Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or
- Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.0 ADMINISTRATIVE CONTROLS

6.4 DELETED

6.5 DELETED

l. Steam Generator (SG) Program (continued)

1. (continued)

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage

PAGES 6-15g AND 6-15h HAVE BEEN DELETED.
THE NEXT PAGE IS 6-15i.

m. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACS, operating at the flow rate required by the VFTP, at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 36 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

n. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(i) Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,
2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
3. A clear and bright appearance with proper color or a water and sediment content within limits;

(ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

(iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

o. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14 , Revision 1, August 1975.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 210 AND 159

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

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNITS NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By application dated December 14, 2009, and supplement dated July 30, 2010, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093580093 and ML102150172, respectively), Florida Power & Light Company (FPL, the licensee) requested changes to the Technical Specifications (TSs) for St. Lucie Plant Unit 1 (SL1) and Unit 2 (SL2). The supplement dated July 30, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 20, 2010 (75 FR 20638). The proposed changes delete the structural integrity TSs, update accident monitoring instrumentation requirements and make various administrative TS changes. Specifically FPL proposes:

1. Deletion of TS 3/4.4.10, "Structural Integrity" (SL1), and TS 3/4.4.11 (SL2);

2. Relocate the SL2 reactor coolant pump flywheel inspection surveillance requirement (SR) 4.4.11 to a new administrative control program 6.8.4.o;

3. Delete TS 6.4.1, "Training," to be consistent with NUREG-1432, Revision 3.0, "Standard Technical Specifications [STSs] Combustion Engineering Plants" (SL1 and SL2) and relocate the definition of licensed operators from TS 6.3.1 to new TS 6.3.2;

4. Revise Accident Monitoring Instrumentation actions 1, 2, 6, and 7 in SL1 TS 3.3.3.8 and Accident Monitoring Instrumentation actions a and b in SL2 TS 3.3.3.6;

5. Change the action in SL2 TS 3.1.2.6 to be logically correct;

6. Correct a typographical error in SL2 SR 4.3.3.2;

7. Correct section heading on index page VI of SL1 TSs;

8. Delete reference to Table 3.6-1 on index page XXIV of SL2 TSs; and

9. Delete TS 6.8.4.I.2 and 6.9.1.13 steam generator inspection requirements SL2 TS.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 establishes the regulatory requirements for TS content. The requirements emphasize preventing accidents and mitigating accident consequences. Applicants are expected to incorporate into their TSs "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity" (see Statement of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports," of December 17, 1968 (33 FR 18610)). Section 50.36 of 10 CFR requires that TSs include items in the following specific categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. However, the rule does not specify the format and content for TS categories.

IMPROVED STANDARD TECHNICAL SPECIFICATIONS PROGRAM

Interim Policy Statement on TS Improvements

The Nuclear Regulatory Commission (NRC, Commission) and industry representatives sought to develop guidelines for improving nuclear power plant TS content and quality. On February 6, 1987, the Commission issued an "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). In September 1992, the Commission issued NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," (STSs), which was developed using the guidance and criteria contained in the Commission's Interim Policy Statement. The STSs are a model for developing improved TSs (ITSs) for Combustion Engineering plants. The Interim Policy Statement criteria ensure that ITSs would consistently reflect system configurations and operating characteristics for the Combustion Engineering design. In addition, the generic Bases statements provide the basis for each of the STS requirements.

Final Policy Statement on TS Improvements

On July 22, 1993, the Commission issued its Final Policy Statement indicating that satisfying the guidance in the policy statement also satisfies Section 182a of the Atomic Energy Act and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement described the improved STSs safety benefits and encouraged licensees to use the improved STSs as the basis for plant-specific TS amendments and for complete conversions to the improved STSs. Further, the Final Policy Statement gave guidance for evaluating the required scope of the ITSs and defined the guidance criteria for determining which of the LCOs and associated surveillances should remain in the ITSs. Using this approach, licensees should keep existing LCO requirements that fall within or satisfy any of the Final Policy Statement criteria in the TSs. Those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995).

Final Policy Statement Criteria

The Final Policy Statement criteria are as follows:

Criterion 1 — Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 — A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to fission product barrier integrity.

Criterion 3 — A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to fission product barrier integrity.

Criterion 4 — A SSC which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

Section 50.36(c)2 of 10 CFR states, "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shutdown the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

Section 50.36(c)3 of 10 CFR states that TS surveillance requirements, "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

The NRC's guidance for the format and content of licensee TS can be found in NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," Revision 3.0.

3.0 TECHNICAL EVALUATION

3.1 Deletion of Structural Integrity TS 3/4.4.10 (SL1) and TS 3/4.4.11 (SL2) and the associated TS Bases

The licensee, in its application, stated that the purpose of TS 3/4.4.10 (SL1) and TS 3/4.4.11 (SL2), structural integrity LCO, is to specify the requirements for maintaining the structural integrity of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, 2, and 3 components. This specification was originally intended to support assurance that structural integrity and operational readiness of these components are maintained at an acceptable level throughout the life of the facility. The specification is applicable in all operational modes. However, the specification does not provide actions for plant shutdown if its LCO is not met. This is because the specification addresses the passive pressure boundary function of ASME Code Class 1, 2, and 3 components as established by compliance with the Inservice Inspection (ISI) program. In addition to the above, the ISI program is required pursuant to 10 CFR 50.55a, "Codes and Standards," and SR 4.0.5 thereby addressing the inspections necessary to maintain structural integrity.

Furthermore, the specification wording could be misconstrued to conflict with normal outage-related activities in preparation for refueling, which would make the reactor coolant system (RCS) pressure boundary no longer structurally intact. The licensee states that maintaining a program-type requirement within an LCO creates significant interpretation issues for Operations personnel. The RCS structural integrity TS was part of the original TSs and, the TS basis history regarding its intent is not documented. TS 3/4.4.10 and TS 3/4.4.11 appear to have been included to help ensure that plant heatup and startup would not occur until all required portions of applicable systems were verified to meet ISI acceptance criteria following inspections performed during a plant outage. Meeting these acceptance criteria helps ensure the integrity of all applicable systems during all modes of operation, including accident events. Furthermore, TS 3/4.4.10 and TS 3/4.4.11 contain no actions suggesting that they were designed to accommodate integrity concerns once plant heatup has commenced. Structural integrity ISI activities are performed only during plant outages when conditions exist that permit access to the applicable systems and are not monitored or controlled through application of the ISI program during the operational cycle.

The licensee stated that other TSs are designed to monitor the structural integrity of the RCS during operation and provide actions to shut down the unit if compliance is not maintained. For example, RCS heatup and cooldown rates TS, and the overpressure mitigation system TS protect against applying undue stresses on RCS components and piping as a result of pressure/temperature transients. The RCS leakage TS provide a means of protecting the RCS integrity by detecting and monitoring leakage. Therefore, the licensee stated it is not necessary to apply TS 3/4.4.10 and TS 3/4.4.11 when integrity issues become evident during plant operation above cold shutdown. Because TS 3/4.4.10 and TS 3/4.4.11 are redundant to other regulations, it is acceptable to remove TS 3/4.4.10 and TS 3/4.4.11 from the TSs. Finally, removal of this specification does not reduce the controls that are necessary to ensure compliance with the ASME Code. Structural integrity is maintained by compliance with 10 CFR 50.55a, as implemented through the St. Lucie Plant ISI Program required by 4.0.5, as well as by compliance with TS 3.4.6.1, 3.4.6.2, 3.4.9.1, 3.4.9.2 and 3.4.9.3 for the RCS.

For Criterion 1, above, the RCS ASME Code Class 1, 2, and 3 components do not include any instrumentation used to detect, and indicate in the control room, a significant abnormal degradation of the RCS. Therefore, the NRC staff finds that TS 3/4.4.10 and TS 3/4.4.11 do not meet Criterion 1.

For Criterion 2, TS 3/4.4.10 and TS 3/4.4.11, "Structural Integrity," are not applicable to a process variable, design feature, or operating restriction that is an initial condition of a design-basis analysis (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Structural integrity is not monitored or controlled during plant operation; it is verified through implementation of the St. Lucie ISI program required by TS 4.0.5. Therefore, the NRC staff finds that TS 3/4.4.10 and TS 3/4.4.11 do not meet Criterion 2.

For Criterion 3, no specific TS related SSCs are being revised or removed from the TS that are part of the primary success path and function or actuate to mitigate DBAs or transients that either assume the failure of, or present a challenge to, the integrity of a fission product barrier. Each TS SSC must continue to meet the requirements of 10 CFR 50.55a as implemented through the St. Lucie ISI program required by TS 4.0.5. Therefore, the NRC staff finds that TS 3/4.4.10 and TS 3/4.4.11 do not meet Criterion 3.

For Criterion 4, the requirements covered by TS 3/4.4.10 and TS 3/4.4.11, that are being removed, have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. In addition, failure modes of applicable SSCs are not identified from the requirements of these TSs. Furthermore, the requirements of these TSs do not affect the risk review/unavailability monitoring of applicable SSCs. Therefore, the NRC staff finds that these specifications do not meet Criterion 4.

The review for the structural integrity LCO relocation was actually performed and presented in a letter from the Director of the Office of Nuclear Reactor Regulation, NRC, Thomas E. Murley, on May 9, 1988. Originally, NUREG-0212, "Standard Technical Specifications for Combustion Engineering Plants," contained provisions for the LCOs and SRs in reference to the structural integrity of ASME Code Class 1, 2, and 3 components. This letter identified Section 3/4.4.10, "Structural Integrity," as not meeting the criterion for 10 CFR 50.36 and, therefore, was removed from subsequent revisions of the STSs.

Therefore, since these TSs do not fulfill any of the 10 CFR 50.36(c)(2)(ii) criteria for items for which TSs must be established, the NRC staff finds that removing TS 3/4.4.10 and TS 3/4.4.11 along with the associated TS bases is acceptable. Finally, the removal of TS 3/4.4.10 and TS 3/4.4.11 and their associated references to structural integrity eliminates from the TSs the redundancy of structural integrity requirements that are already covered under 10 CFR 50.55a.

Normally in applying the Commission Final Policy Statement on Technical Specifications for Nuclear Power Reactors, the NRC staff would require that a licensee identify both the licensee-controlled document receiving a relocated TS and the change control mechanism that governs that document. However, in this instance to achieve efficiency in the issuance of this license amendment to relocate TS 3/4.4.10 and TS 3/4.4.11, elimination of a duplicate regulatory requirement, the NRC staff will permit deletion without relocation of the TSs. Therefore, the NRC staff finds this proposed change to be acceptable.

3.2 <u>Relocate the SL2 reactor coolant pump flywheel inspection surveillance requirement (SR) 4.4.11 to a new administrative control program 6.8.4.o</u>

The licensee, in its application, stated that the reactor coolant pump (RCP) flywheel inspection requirement located SR 4.4.11 in SL2 TS would be relocated to new Administrative Controls program 6.8.4.o. The wording will be reworded to be identical to the RCP flywheel inspection program requirements in NUREG-1432, Revision 3.0, TS 5.5.7, and the prescribed inspection methods will not be revised from the current requirements.

SL2 SR 4.4.11 states, "In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide (RG) 1.14, Revision 1, August 1975."

NUREG-1432, Revision 3.0, Administrative Control program TS 5.5.7 states that the program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position C.4.b of RG 1.14, Revision 1, August 1975.

The NRC staff finds this proposed change to be administrative and acceptable since the inspection requirements remain unchanged and are only being relocated.

3.3 Delete TS 6.4.1, "Training," to be consistent with NUREG-1432, Revision 3.0, "Standard Technical Specifications Combustion Engineering Plants" (SL1 and SL2)

SL1 and SL2 Updated Final Safety Analysis Report (FSAR) Section 13.2, describes the training program as meeting or exceeding the requirements and recommendations of Section 5.5 of American National Standards Institute (ANSI)/American Nuclear Society (ANS)-3.1 1978 and 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980, NRC letter to all licensees as outlined in Section 6.4, training, of the plant TSs.

The licensee has proposed to delete TS 6.4.1 in SL1 and SL2 and to relocate the definition of licensed operators from TS 6.3.1 to new TS 6.3.2 to be consistent with the NUREG-1432, Revision 3.0, wording for unit staff qualifications, TS 5.3. NUREG-1432, Revision 3.0, TS 5.3 states:

> 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. [The staff not covered by Regulatory guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff.]"
>
> 5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specifications 5.3.1, perform the functions described in 10 CFR 50.54(m).

On March 19, 1987, Generic Letter (GL) 87-07, "Information Transmittal of Final Rulemaking for Revisions to Operator Licensing - 10 CFR Part 55 and Conforming Amendments," informed facility licensees that they had the option of substituting an accredited, Systems Approach to Training based program for their operator training program previously approved by the NRC. The GL indicated that this option may be implemented upon written notification to the NRC and that it did not require any staff review. The GL also noted the NRC's expectation that facility licensees would update their licensing basis documents (e.g., their FSARs and TSs), as necessary, to conform to their accredited program status.

As stated in Regulatory Information Summary (RIS) 2001-001, the NRC has not changed its requirements or position with regard to license eligibility for ROs and SROs since 1987. RG 1.8, Revision 3 and the National Academy for Nuclear Training guidelines for education and experience outline acceptable methods for implementing the Commission's regulations in this area. As stated in RIS 2001-001, any required TS changes would be considered administrative in nature.

The NRC staff finds this proposed change to be administrative and, therefore, acceptable.

3.4 <u>Revise Accident Monitoring Instrumentation actions 1, 2, 6, and 7 in SL1 TS 3.3.3.8 and Accident Monitoring Instrumentation actions a and b in SL2 TS 3.3.3.6</u>

The licensee, in its application, stated that the primary purpose of the postaccident monitoring instrumentation is to display plant variables that provide information required by the control room operators during accidents. The information provides the necessary support for the operators to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for design basis events.

Furthermore, SL1 TS 3.3.3.8 Table 3.3-11 actions 1 and 2 have a specified end state of hot standby. The required end state where the LCO is no longer applicable is hot shutdown. Consistent with NUREG-1432 STS for postaccident monitoring instrumentation, the proposed change drives the action end state to hot shutdown within 12 hours through the intervening state of hot standby within 6 hours. The proposed hot standby completion times are equal to or more conservative than the existing mode 3 completion times and are acceptable. The hot shutdown end state and completion times are equal to or more conservative than the existing end state and completion times and are acceptable. They are also consistent with NUREG-1432, Revision 3, requirements. SL1 TS 3.3.3.8 Table 3.3-11 actions 6 and 7, and SL2 TS 3.3.3.6 actions a and b have a specified end state of hot shutdown with a completion time of 12 hours. The completion time, 12 hours, is consistent with the NUREG-1432, Revision 3, completion time for entry into mode 4 and is acceptable. FPL proposes to include the 6-hour completion time to enter the intervening state of hot standby. This change is consistent with the NUREG-1432, Revision 3, completion time for entry into mode 3 conditions.

SL1 TS 3.3.3.8, "Accident Monitoring Instrumentation" and SL2 TS 3.3.3.6, "Accident Monitoring Instrumentation" LCOs are both applicable in modes 1, 2, and 3. SL1 and SL2 TS define mode 1 as power operation, mode 2 as startup, mode 3 as hot standby and mode 4 as hot shutdown. TS 3.3.3.8 action 1 requires restoring inoperable accident monitoring instrumentation to operable status within the specified completion time or requires placing the plant in hot standby within the next 12 hours. TS 3.3.3.8 action 2 requires restoring inoperable accident monitoring instrumentation to operable status within the specified completion time or requires placing the plant in hot standby within the next 6 hours. TS 3.3.3.8 actions 6 and 7, and TS 3.3.3.6 actions a and b require restoring inoperable accident monitoring instrumentation to operable status within the specified completion time or requires placing the plant in hot shutdown within the next 12 hours. The licensee's proposed change will require the plant to be in hot standby in 6 hours and hot shutdown in 12 hours if the inoperable accident monitoring instrumentation cannot be returned to operable status within the specified completion time for TS 3.3.3.8 actions 1, 2, 6, and 7, and TS 3.3.3.6 actions a and b.

This change requires: (1) SL1 to be in hot standby in 6 hours, which is more restrictive than the allowed 12 hours in action 1 of TS 3.3.3.8, (2) SL1 to continue to shutdown the plant to mode 4 (hot shutdown), which is more restrictive than the allowed mode 3 (hot standby) in action 1 and 2 of TS 3.3.3.8 and (3) SL1 and SL2 to be in hot standby in 6 hours prior to the current TS requirement to be in hot shutdown in 12 hours in actions 6 and 7 of TS 3.3.3.8 and actions a and b of TS 3.3.3.6.

The NRC staff finds this proposed change to be more restrictive or equivalent to the current specified completion time requirements in SL1 and SL2 TSs, and is consistent with NUREG-1432, Revision 3. This proposed change is acceptable.

3.5 Change the action in SL2 TS 3.1.2.6 to be logically correct

SL2 TS 3.1.2.6, "Boric Acid Makeup Pumps – Operating" required action states:

> With no boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to its COLR [core operating limits report] limit at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

The licensee, in its application, stated that changing the required action to read "With no boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 *operable*..." will make the required action be logically consistent.

The NRC staff agrees with this conclusion and finds this change acceptable.

3.6 Correct a typographical error in SL2 SR 4.3.3.2

SL2 TS 3.3.3.1, "Radiation Monitoring Instrumentation" SR 4.3.3.2 states, "At lease once per 18 months, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits."

The licensee proposes to correct a typographical error by changing the SR to read "at *least* once per 18 months..."

The NRC staff agrees that this change corrects a typographical error. The NRC staff finds this change to be administrative and acceptable.

3.7 Correct section heading on index page VI of SL1 TSs

SL1 TS index page VI lists the section heading for containment systems as, "3/4.4.6 CONTAINMENT SYSTEMS." The licensee proposes to correct the section heading by changing the section heading to "3/4.6 CONTAINMENT SYSTEMS." The NRC staff agrees with this change. The NRC staff finds this change to be administrative and acceptable.

3.8 Delete reference to Table 3.6-1 on index page XXIV of SL2 TSs

SL2 TS index page XXIV list of tables includes, "Table 3.6-1 Containment leakage paths." The licensee proposes to delete the reference to Table 3.6-1, "Containment leakage paths." The licensee, in its application, stated that Table 3.6-1 was removed from the TS in Amendment 88, which implemented 10 CFR 50 Appendix J, Option B.

The NRC staff agrees that the containment leakage path Table 3.6-1 was removed from SL2 TS in Amendment 88. The NRC staff finds this change to be administrative and acceptable.

3.9 Delete TS 6.8.4.I.2 and 6.9.1.13 steam generator inspection requirements SL2 TS

The licensee, in its application, stated that SL2 TSs 6.8.4.I.2 and 6.9.1.13 pertain to the steam generator integrity program and reporting requirements for the SL2 original steam generators and are no longer applicable to the replacement steam generators that were installed in Refueling Outage 17 (SL2-17). The replacement steam generator inspection/reporting requirements are unchanged and are contained in TSs 6.8.4.I.1 and 6.9.1.12.

SL2 TS 6.8.4.I.2 states, "A SG [steam generator] program shall be established and implemented for the original SGs to ensure that SG tube integrity is maintained. In addition, the SG program shall include the following provisions: . . ."

SL2 TS 6.9.1.13 states, "A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection of the original SGs performed in accordance with Specifications 6.8.4.I.2. The report shall include: . . ."

The NRC staff agrees with the licensee's conclusion. The replacement steam generator inspection/reporting requirements are unchanged and contained in TSs 6.8.4.I.1 and 6.9.1.12. Both TSs 6.8.4.I.2 and 6.9.1.13 provide requirements for the original steam generators for SL2. Since the SL2 steam generators were replaced in the SL2-17 refueling outage, TSs 6.8.4.I.2 and 6.9.1.13 are no longer applicable. Therefore, the NRC staff finds this change to be administrative and acceptable.

4.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, 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. 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 (75 FR 20638, dated April 20, 2010). 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.

Principal Contributor: Kristy Bucholtz

Date: December 28, 2010

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/

Tracy J. Orf, 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. 210 to DPR-67
2. Amendment No. 159 to NPF-16
3. Safety Evaluation

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