April 18, 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 UNIT 2 - ISSUANCE OF AMENDMENT REGARDING REDUCTION IN MINIMUM REACTOR COOLANT SYSTEM FLOW (TAC NO. MB6526)

Dear Mr. Stall:

The Commission has issued the enclosed Amendment No. 131 to Facility Operating License No. NPF-16 for the Saint Lucie Plant, Unit 2. These amendments consist of changes to the Technical Specifications in response to your application October 15, 2002, as supplemented on February 28, 2003.

This amendment decreases the value of design reactor coolant system flow rate. This reduction in the reactor coolant system flow requirement will support operation of the plant with an increased steam generator (SG) tube plugging level to a maximum of 1250 tubes per SG.

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 No. 50-389

Enclosures:
1. Amendment No. 131 to NPF-16
2. Safety Evaluation

cc w/encls: See next page

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*See previous concurrence

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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. 131
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), October 15, 2002, as supplemented February 28, 2003, 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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No.131, 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 18, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 131

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

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 Page	Insert Page
2-5	2-5
3/4 2-15	3/4 2-15

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 131 TO FACILITY OPERATING LICENSES NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

SAINT LUCIE PLANT, UNIT 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated October 15, 2002 (Reference 1), as supplemented on February 28, 2003 (Reference 2), Florida Power and Light Company (FPL, the licensee) requested approval for a Technical Specification (TS) change for Saint Lucie Unit 2 (SL2). The licensee requested a change to TS Table 3.2-2 and the footnote to Table 2.2-1 to allow a reduction of the reactor coolant system (RCS) flow rate from 363,000 gallons per minute (gpm) to 355,000 gpm. This reduction was proposed to accommodate a steam generator (SG) tube plugging increase to 1250 tubes per SG, which is already supported by a current safety analysis. However, the analysis was performed based on an RCS flow rate of 363,000 gpm.

Additionally, the wording of the Bases for TSs 2.1.1, 2.2.1 and 3/4.2.5 would be changed to replace "DNB-SAFDL [departure from nucleate boiling-specified acceptable fuel design limit] of 1.28" with "appropriate correlation limit for DNB-SAFDL." This change was proposed since there is more than one departure from nuclear boiling ratio (DNBR) correlation approved for SL2, namely, CE-1, ABB-NV and MacBeth for low-flow conditions. The licensee currently uses CE-1 and MacBeth, but can use the other correlation if additional margin is needed due to additional SG tube plugging.

The licensee's supplement dated February 28, 2003, did not affect the original proposed no significant hazards determination, nor expand the scope of the request as noticed in the *Federal Register* on November 12, 2002 (67 FR 68737).

2.0 REGULATORY EVALUATION

The proposed amendment for the change of minimum RCS flow rate must comply with the acceptance criteria of Title 10, *Code of Federal Regulations (*10 CFR) Section 50.46, and Criterion 10 of 10 CFR Part 50, Appendix A, General Design Criteria for Nuclear Power Plants. Standard Review Plan Chapters 4.4 and 15 provide guidance for confirming that these regulatory requirements are met. The licensee is required to demonstrate that the RCS with the reduced minimum RCS flow rate will provide appropriate margin to assure that the SAFDL are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs) and Loss of Coolant Accidents (LOCA).

Enclosure 2

3.0 TECHNICAL EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in References 1 and 2.

The licensee assessed the impact of reduced RCS flow on Peak Clad Temperature (PCT), Maximum Core-Wide Cladding Oxidation (MCWO), and Maximum Cladding Oxidation (MCO) based on a 355,000 gpm flow rate during a Large Break LOCA. The reduced RCS flow results in an increase in PCT of 15°F and an MCO increase of 0.40 percent. There was no increase in MCWO. The NRC staff verified that the reduced flow case PCT, cladding oxidation and hydrogen generation remained below the 10 CFR 50.46 acceptance criterion.

The licensee calculated the impact of reduced RCS flow on the Small Break LOCA using input consistent with 355,000 gpm. The reduced RCS flow resulted in an increase in PCT of 7.6°F and an MCO increase of 0.16 percent and an MCWO increase of 0.02 percent. The NRC staff verified that the PCT, hydrogen generation, and cladding oxidation remained below the 10 CFR 50.46 acceptance criteria.

The licensee performed the decay heat removal part of the Long Term Cooling (LTC) analysis with inputs consistent with 355,000 gpm. The results indicated that the boric acid precipitation part of the LTC analysis was not changed by the changes in operating conditions. Therefore, the reduction of the RCS flow does not affect the post-LOCA LTC results.

The licensee evaluated a reduction in the RCS flow to determine the impact on the containment pressure and temperature. The evaluation concluded that the flow reduction has an insignificant impact on the containment pressure and temperature, as well as the Component Cooling Water/Intake Cooling Water system temperature response.

The NRC staff previously approved the analysis methods used by the licensee for the design basis analysis of transients and accidents, and the criteria applied to each event. In the safety evaluation supporting Amendment 105 to the SL2 license (Ref. 3), the NRC staff stated "Since the methods used in the non-LOCA transient analyses were previously approved by the NRC and the design parameters for the SL2 reactor core are within the applicable ranges of the approved methods, the NRC staff concludes that the application of the previously approved methods is acceptable for transient analyses to support the SL2 reload applications."

The non-LOCA safety analyses included discussion of whether the appropriate acceptance criteria were met. The licensee performed the analyses based on 2700 MWt, 355,000 gpm minimum reactor vessel flow, and a reduction of the inlet temperature by 1°F to 549°F, consistent with the TS limit value.

With regard to the DNB correlations available for use, the NRC staff questioned how the licensee chose which correlation to use, whether the correlations were approved for use with the current fuel, and whether more than one correlation needed to be used for a cycle. Both the CE-1 and ABB-NV correlations are approved for the fuel, and each correlation is applicable to all of the fuel in the core. The licensee decides which correlation to use based on the margin needed due to plant configuration changes, such as steam generator tube plugging. This is typically done through the reload analysis process. The licensee stated that only the CE-1 correlation is currently used. Based on the above discussion, the NRC staff finds the use of the different correlations and the licensee's process for selecting which correlation to use acceptable.

The overpower margin (OPM) is the percent power which results in the DNBR limit being reached with the other relevant operating parameters at the steady state operating conditions. The Required Overpower Margin (ROPM) is defined in CENPD-199-P, "C-E Setpoint Methodology," for loss of flow events and for symmetric steam generator transients events. The ROPM for DNB is the minimum OPM required to meet the DNBR limit with the relevant parameters at the limiting conditions and appropriate reactor protection systems setpoints. Operating with available over power margin (AOPM) greater than or equal to the ROPM ensures that the DNB-SAFDL will be met for all anticipated operational occurrences. The NRC staff questioned how the SAFDL would be met for all non-LOCA accident events with quantified results. For DNB-SAFDL, the licensee evaluated the effects of the reduced RCS flow rate on the ROPM for various transients, and provided the effects of the reduced RCS flow on the ROPM for various transients. The maximum effect was shown to increase 1.6 percent on the ROPM for the pre-trip steam line failure. The licensee also stated that plant setpoints reserve more margin than the transient analysis ROPM by at least 5 percent, which bounds the 1.6 percent ROPM increase. The licensee re-evaluated various events which challenge the DNBR limit. The evaluation determined that the consequences of the events were unchanged from the Updated Final Safety Analysis Report conclusions for all cases, and the DNBR was preserved. Therefore, the NRC staff finds that the DNB-SAFDL is satisfied.

With regard to maximum clad surface temperature during normal operation, the NRC staff questioned why the temperature didn't increase due to reduced RCS flow. FPL's response (Ref. 2) was that they use the Jens-Lottes correlation. Since the RCS flow reduction is not a dependent variable for this correlation, the maximum clad surface temperature did not change. The NRC staff finds this response acceptable.

The NRC staff asked for clarification of the plot of axial power distribution on page 3 of Attachment 1 of the original submittal. The licensee provided the required clarification and stated that the curves were axial power distributions with corresponding radial peaks.

The licensee evaluated or analyzed several events to assure that peak pressure criteria were met. The criterion is 110 percent of the RCS design pressure (i.e., 2750 psia for AOOs, and 3000 psia for the low probability Feedwater Line Break Event). The Loss of Condenser Vacuum Event is the bounding AOO case and meets the 2750 psia criterion by 2.9 psia. The very low probability Feedwater Line Breaks with a Loss of AC power meets the 3000 psia criterion by a margin of 153 psi. Based on the bounding AOO case meeting the peak pressure criteria, the NRC staff finds the change in peak pressure acceptable.

The licensee analyzed the inadvertent opening of a steam generator safety valve or atmospheric dump valve and the post-trip steam system piping failures for shutdown margin. The licensee indicated that the shutdown margin associated with the inadvertent opening of a SG safety valve or atmospheric dump valve event was insensitive to flow. The results of the analysis, indicated that the maximum post-trip reactivity decreased from 0.4407 percent $\Delta\rho$ to 0.4373 percent $\Delta\rho$. Based on a decrease in the maximum post-trip reactivity, the NRC staff finds the change in shutdown margin acceptable.

Based on the changes, including the maximum clad surface temperature, peak pressure, and shutdown margin, maintaining an appropriate margin to assure that the SAFDL would not be exceeded, the NRC staff concludes that the change to TS Table 3.2-2 and the footnote to Table 2.2-1 to allow a reduction of the RCS flow rate from 363,000 gpm to 355,000 gpm is acceptable. Also, the NRC staff acknowledges the change to the wording of the Bases for TSs 2.1.1, 2.2.1, and 3/4.2.5 to replace “DNB-SAFDL of 1.28" with “appropriate correlation limit for DNB-SAFDL” appears consistent with the TS change.

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, NRC, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 amendment involves 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 amendment involves no significant hazards consideration and there has been no public comment on such finding (67 FR 68737, dated November 12, 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) Letter from D.E. Jernigan (FPL) to NRC, “Reduce the Minimum Reactor Coolant System Flow,” October 15, 2002.

2) Letter from D.E. Jernigan (FPL) to NRC, “Reduce the Minimum Reactor Coolant System Flow ,” February 28, 2003.

3) Letter to T.F. Plunkett (FPL) from K.N. Jabbour (NRC), “St. Lucie Unit 2 - Issuance of Amendment Regarding the Cycle 12 Reload Process Improvement (TAC No. MA452),” December 21, 1999 (amended in letter dated March 1, 2000).

4) Letter to D.E. Jernigan (FPL) from B.T. Moroney (NRC), “St. Lucie Plant, Unit 2 - Request for Additional Information Regarding Reduction in Minimum Reactor Coolant System Flow,” January 30, 2003.

Principal Contributor: Sarah E. Colpo, NRR

Date: April 18, 2003