May 20, 1987

Docket No. 50-335

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Mr. C. O. Woody
Group Vice President
Nuclear Energy
Florida Power & Light Company
P. O. Box 14000
Juno Beach, Florida 33408

Dear Mr. Woody:

The Commission has issued the enclosed Amendment No. 80 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated December 18, 1986.

This amendment deletes the technical specifications associated with core support barrel excessive movement (TS 3/4.4.11). Core support barrel movement has been monitored for over nine years and no excessive motion has been detected.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/s/

E. G. Tourigny, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II

Enclosures:
1. Amendment No. 80 to DPR-67
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDII-2	PD:PDII-2	D:PDII-2	OGC
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Mr. C. O. Woody
Florida Power & Light Company

St. Lucie Plant

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
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 18, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 80 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L S Rubenstein

Lester S. Rubenstein, Director
Project Directorate II-2
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: May 20, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 80

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
V	V
X	X
3/4 4-56	3/4 4-56
3/4 4-57	3/4 4-57
B3/4 4-13	B3/4 4-13

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

SECTION	PAGE

3/4.4 REACTOR COOLANT SYSTEM

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.6 CONTAINMENT SYSTEMS

TABLE 4.4-8

INSERVICE INSPECTION PROGRAM - SAFETY CLASS 3 COMPONENTS

1. Class 3 piping greater than 4 inches and components with pipe connections greater than 4 inches will be pressure tested and visually examined near or at the end of the 10-year inspection interval. In addition, components will be visually examined during periods of normal operations or during system performance testing once during each 1/3 of the year 10-year inspection interval.

2. Open-ended portions of systems will not be pressure tested.

3.4.11 DELETED

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The nondestructive testing for repairs on components greater than 2 inches diameter gives a high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. Repairs on components 2 inches in diameter or smaller receive a surface examination which assures a similar standard of integrity. In each case, the leak test will ensure leak tightness during normal operation.

For normal opening and reclosing, the structural integrity of the Reactor Coolant System is unchanged. Therefore, satisfactory performance of a system leak test at 2235 psia following each opening and subsequent reclosing is acceptable demonstration of the system's structural integrity. These leak tests will be conducted within the pressure-temperature limitations for Inservice Leak and Hydrostatic Testing and Figure 3.4-2.

The Safety Class 2 and 3 components will be pressure tested at least once toward the end of each inspection interval (10 years). The Safety Class 2 components having a design temperature above 400°F will be pressure tested at not less than 125 percent of the system design pressure while those components having a design temperature of 400°F and below will be pressure tested at 110 percent of design pressure. The Safety Class 3 components will be pressure tested at the levels indicated in Specification 4.4.10.3b.

3/4.4.12 PORV BLOCK VALVES

The opening of the Power Operated Relief Valves fulfills no safety related function. The electronic controls of the PORVs must be maintained OPERABLE to ensure satisfaction of Specifications 4.5.1.d.1 and 4.5.2.d.1. Since it is impractical and undesirable to actually open the PORVs to demonstrate reclosing, it becomes necessary to verify operability of the PORV Block Valves to ensure the capability to isolate a malfunctioning PORV.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 80

TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

INTRODUCTION

By letter dated December 18, 1986, Florida Power & Light Company (FP&L) proposed an amendment to delete the Core Barrel Movement Technical Specifications 3.4.11, 4.4.11.1 through 4.4.11.3, and Bases 3/4.4.11. The purpose of the Technical Specifications was to verify the effectiveness of the redesigned core barrel hold-down ring by determining the core barrel movement baseline and by monitoring core barrel movement against the baseline.

During licensing of St. Lucie Unit 1, a problem was identified at Palisades and several other Combustion Engineering reactors, including St. Lucie Unit 1, concerning the core barrel hold-down ring design. NRC addressed this problem in Section 3.9.1 of the St. Lucie Unit 1 Safety Evaluation Report (SER) dated November 8, 1974, and stated in the SER, "A monitoring program will be required until either a modification has been made to the internals or data indicates the program may be discontinued." The St. Lucie Unit 1 core barrel hold-down ring was redesigned to provide additional force to hold the core barrel in place, and in supplement 1 to the SER, NRC stated that the redesigned ring was acceptable and that the issue was resolved with incorporation of a surveillance program to monitor core barrel movement.

This NRC position was reflected in the Bases for the Core Barrel Movement Technical Specification in the statement, "A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of the first fuel cycle."

Based on a review of the existing data, FP&L believes that an adequate basis has been provided to justify deletion of the Technical Specifications.

DISCUSSION

By letter dated April 22, 1977 (L-77-122), FP&L submitted to NRC the Core Barrel Movement Baseline Report as required by Technical Specification 4.4.11.1. The baseline was established by monitoring core support barrel motion at nominal power levels of 20, 50, 80, and 100 percent of rated thermal power during the reactor startup test program. As stated in the report, the core support barrel is moving less than + 8.8 mils in amplitude (99.7 percent confidence level) at the snubber gap level.

The baseline monitoring results provided sufficient verification of the effectiveness of the redesigned core barrel hold-down ring, in that Palisades had experienced approximately 300 mils amplitude motion as determined from the measured wear of the snubber blocks. However, St. Lucie Unit 1 has continued monitoring core barrel movement in accordance with the Technical Specifications. Results of the monitoring program have been included in the Annual Operating Reports, beginning in the 1977 report. Based on a review of the results presented in these nine reports, it is seen that the core barrel motion has been as expected. Furthermore, upon identification of the thermal shield problem in Spring 1983, the core barrel was removed, inspected and, where damaged, repaired. During the post repair inspection, all six snubber blocks were examined and there were no indications of excessive core barrel movement.

SUMMARY

The redesigned core barrel hold-down ring has eliminated the possibility of excessive core barrel movement such as that occurred at Palisades. This has been verified by more than nine years of core barrel movement monitoring and by physical inspection of the core barrel snubber blocks.

The staff concurs with the FP&L assessment that the purpose of the Technical Specifications has been satisfied and an adequate basis has been provided to justify deletion of the Technical Specifications 3.4.11, 4.4.11.1 through 4.4.11.3, and Bases 3/4.4.11 relative to the Core Barrel Movement.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation, use or surveillance of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental statement or environmental assessment need be prepared in connection with the issuance of the amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 20, 1987

Principal Contributor:
J. Rajan