Docket No. 50-389 May 23, 1991

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 UNIT 2 - ISSUANCE OF AMENDMENT RE: RESISTANCE TEMPERATURE DETECTOR DELAY TIME (TAC NO. 69863)

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application dated October 24, 1988, as supplemented June 1, 1989, October 19, 1989, March 27, 1990, November 8, 1990 and modified December 18, 1990.

This amendment changes the maximum allowable primary loop resistance temperature detection delay time from 8 seconds to 14 seconds.

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. 50 to NPF-16
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD22	PM:PD22	D:PD22	OGC			
NAME	DMiller	JNorris/jkd	HBerkow	Sutton			
DATE	5/1/91	5/2/91	5/2/91	5/7/91			

on condition that SER be amended

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Document Name: AMEND ST. LUCIE 69863

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DATED: May 23, 1991

AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. NPF-16 - ST. LUCIE, UNIT 2

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

St. Lucie Plant

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

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. 50
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 24, 1988, as supplemented June 1, 1989, October 19, 1989, March 27, 1990, November 8, 1990 and modified December 18, 1990, 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. 50 , 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

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: May 23, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 50

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 areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
B 2-4	B 2-4
3/4 3-7	3/4 3-7

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1971 Edition including Addenda to the Summer, 1973, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Power Level-High

A Reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure Trip.

The Variable Power Level High trip setpoint is operator adjustable and can be set no higher than 9.61% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 15.0% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the safety analyses.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2375 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.28.

The trip is initiated whenever the Reactor Coolant System pressure signal drops below either 1900 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 2.0% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 3.0°F to compensate for potential temperature measurement uncertainty; and a further allowance of 125 psia to compensate for pressure measurement error and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 125 psia allowance is made up of a 55 psia pressure measurement allowance and a 70 psia time delay allowance.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

Variable Power Level-High

A Reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure Trip.

The Variable Power Level High trip setpoint is operator adjustable and can be set no higher than 9.61% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 15.0% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the safety analyses.

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TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
10. Loss of Component Cooling Water to Reactor Coolant Pumps	Not Applicable
11. Reactor Protection System Logic	Not Applicable
12. Reactor Trip Breakers	Not Applicable
13. Wide Range Logarithmic Neutron Flux Monitor	Not Applicable
14. Reactor Coolant Flow - Low	0.65 second
15. Loss of Load (Turbine Hydraulic Fluid Pressure - Low)	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Based on a resistance temperature detector (RTD) response time of less than or equal to 14.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 50

TO FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated October 24, 1988, as supplemented June 1, 1989, October 19, 1989, March 27, 1990, November 8, 1990 and modified December 18, 1990, Florida Power and Light Company (FPL, the licensee) proposed to amend Facility Operating License NPF-16 to relax the maximum primary loop resistance temperature detector (RTD) delay time from 8.0 seconds to 16.0 seconds as shown in Table 3.3-2 of the Technical Specifications. This change would provide increased operational flexibility by decreasing problems encountered with replacement of the RTDs. Also, the Technical Specifications (TS) require RTD response time tests to be conducted approximately every 18 months to check the RTDs used for reactor protective instrumentation. The loop current step response (LCSR) time test method is currently used. During the tests, the licensee has encountered difficulties with RTD installation and removal because of the physical location of the support structures which interfere with access to some RTDs. Also, the close proximity of the RTD element to the thermowell increases the potential for galling, which has a negative effect on RTD response time capabilities. Therefore, the licensee desires more flexibility by allowing an increase in allowable response time to decrease the necessity of having to replace the RTDs when they reach the 8.0 second response time limit.

2.0 EVALUATION

As a result of staff's request for additional information, the licensee responded by letters dated June 1, 1989, October 19, 1989, March 27, 1990 and November 8, 1990. The licensee stated that the original accident analysis was based on a total RTD response time of 8 seconds, but that the current accident analysis incorporates a total RTD response time of 16 seconds. Information was presented on the past results of RTD response time tests at the St. Lucie Unit 2 plant in the June 1, 1989 letter. The older RTDs were manufactured by Rouge and de Forest Company (RdF) for which the RTD response time averaged approximately 6.0 seconds (the overall spread was from 2.0 to 7.2 seconds). The largest difference in repeatability at a given location was 1.35 seconds for the RdF RTDs. The new RTDs installed at St. Lucie Unit 2 are manufactured by the Weed Company, for which the RTD response time has averaged a value of 3.0 seconds (the overall spread was 2.0 to 5.0 seconds). The largest difference in drift/repeatability at a given location was shown to be 2.81 seconds for the Weed RTDs. The licensee indicated that they do not have a further analysis or a breakdown of data that determines plant-specific electronic delay or lag time.

The value for the total response time needs to consider other phenomena in addition to the RTD sensor response time alone. Specifically, it is common to experience a 1.0 to 2.0 second delay in response time due to electronic delay. According to NUREG-0809, "Review of Resistance Temperature Detector Time Response Characteristics," August 1981, the LCSR method for response time checking is for field use and is not as accurate as a laboratory test. The LCSR method is accurate only to within 10%. Also, RTDs experience some drift in response time during the 18-month interval between tests. There may also be the repeatability differences mentioned above. In sum, the electronic delay and other uncertainties, such as accuracy of the LCSR method and aging/drift/repeatability, have to be accounted for to obtain the total RTD response time.

The licensee's letter of November 8, 1990 indicated that a new analysis was performed in October 1990 in which an RTD response time of 19.0 seconds was found to be acceptable. The analysis included conservative values for initial core power, coolant inlet temperature, initial reactor coolant system (RCS) pressure, RCS flow, scram worth, moderator temperature coefficient, and core-related parameters. During a telephone conference held on November 28, 1990, the licensee stated that the limiting accident for this analysis was the excess load event. Although the licensee indicated that the margin is estimated to be 3.0 seconds, specific breakdowns of the margin elements were not given. For the TS, the largest acceptable RTD sensor response time can be obtained by working backwards from the 19.0 second analysis value of RTD response time and subtracting time values for LCSR accuracy and aging/electronic drift/repeatability. Using conservative estimates, this amounts to 19.0 - (1.5 + 3.5) = 14.0 seconds. For St. Lucie Unit 2, the Weed RTD response time average is 3.0 seconds, as shown in the data provided by the licensee in the letter dated June 1, 1989. Therefore, the 14.0 second response time allows approximately 11.0 seconds (14.0 - 3.0 seconds) to be available for operational flexibility. The associated Bases pages are also being revised to reflect the change.

In summary, the licensee's original request was for increasing the allowable RTD delay time from 8.0 seconds to 16.0 seconds. Based on the above evaluation, the staff believes that 14.0 second delay time is more appropriate and acceptable. By letter dated December 18, 1990, the licensee agreed with the staff and accordingly modified its request. As a result of this modification of the request, the staff finds the proposed change acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

This 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (53 FR 46146 and 56 FR 6873). Accordingly, this 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 impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: H. Balukjian

Date: May 23, 1991