December 19, 1988

Docket No. 50-389

DISTRIBUTION
See attached sheet

Mr. W. F. Conway
Senior Vice President-Nuclear
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Conway:

SUBJECT: ST. LUCIE UNIT 2 - ISSUANCE OF AMENDMENT RE: CONTROL ELEMENT ASSEMBLY MAXIMUM DROP TIME (TAC NO. 69858)

The Commission has issued the enclosed Amendment No.38 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 20, 1988, as supplemented November 21, 1988.

This amendment changes the control element assembly maximum drop time from 2.7 seconds to 3.1 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,

Herbert N. Berkow FOR

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

Enclosures:
1. Amendment No.38 to NPF-16
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDII-2	PM:PDII-2	NRR/SRXB	OGC-WF	D:PDII
DMiller	ETourigny	WHodges		HBerkow
11/28/88	11/28/88	12/1/88	12/9/88	11/28/88

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cc

Mr. W. F. Conway
Florida Power & Light Company

St. Lucie Plant

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DATED: December 19, 1988

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-16 - ST. LUCIE, UNIT 2

[redacted]
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Others as required

cc: Plant Service list

OF01

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. 38
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 20, 1988, as supplemented November 21, 1988, 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.

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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. 38 , 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: December 19, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 38

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-335

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

Remove Page	Insert Page
3/4 1-24	3/4 1-24

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA position indicator channel shall be OPERABLE for each shutdown or regulating CEA not fully inserted.

APPLICABILITY: MODES 3*, 4,* and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required CEA position indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip breakers in the closed position.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. T_{avg} greater than or equal to 515°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length CEA determined to exceed the above limit:

1. If in MODE 1 or 2, be in at least HOT STANDBY within 6 hours, or

2. If in MODE 3, 4, or 5, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and installation of the reactor vessel head,

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 38

TO FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

INTRODUCTION

By letter dated October 20, 1988, as supplemented November 21, 1988, the Florida Power and Light Company (the licensee) submitted a request to revise Technical Specification 3/4.1.3.4, "CEA Drop Time" for the St. Lucie Plant, Unit No. 2. The change would increase the time requirement for insertion of control element assemblies (CEA's) upon receipt of a reactor trip signal from 2.7 seconds to 3.1 seconds. Specifically, this represents the time from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position from the fully withdrawn position.

A proposed no significant hazards consideration determination was published in the Federal Register on November 16, 1988, based upon the licensee's October 20, 1988 application. However, the application had two pages of the licensee's safety analysis missing. The November 21, 1988 letter forwarded the two pages. Therefore, the November 21, 1988 letter did not affect, in any way, the staff's proposed determination that the amendment request involved a no significant hazards consideration.

NRC Information Notice No. 88-47, "Slower-Than Expected Rod-Drop Time Testing," dated July 14, 1988, advised licensees that drop times could be different depending upon which breakers were used. One CEA drop time could be measured when power is interrupted through the reactor trip breaker, and another drop time could be measured for the same CEA when power is interrupted through the control element drive mechanism (CEDM) circuit breakers. The difference could amount to a few tenths of a second and could affect the operability determination of individual control rods. From a practical point of view, rod drop times measured from when power is interrupted to the reactor trip breakers represent a more realistic scenario under actual reactor trip conditions.

The licensee assumed an increase of 0.4 seconds from the previously assumed 0.34 seconds to 0.74 seconds as a result of an increase in the assumed CEDM holding coil delay time. Large inductance coils around the CEDM magnetically hold the CEA's in position. When a scram signal is received, these holding coils are de-energized. However, because of the large currents passing through these coils, there is a time delay associated with the decay of the magnetic field. After the CEDM holding coil decay delay time, the CEDM's physically disengage and the CEA's drop into the core.

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EVALUATION

The following evaluation addresses an assumed increase in rod drop time of 0.4 seconds and its affect on the licensee's accident and transient analyses of record. The licensee did not provide any data on actual rod drop times, and thus the relationship between an assumed 0.4 second time increase and actual plant performance is not addressed. The 0.4 second increase in rod drop time appears to be a reasonable value from an analysis standpoint. The plant discussed in Information Notice No. 88-47 observed an approximately 0.25 second increase in CEA drop time.

The licensee re-evaluated the design basis events as found in the Unit 2 Updated Final Safety Analysis Report. These events were grouped into the following categories.

1. Increase in heat removal by the secondary system

2. Decrease in heat removal by the secondary system

3. Decrease in reactor coolant flowrate

4. Reactivity and power distribution anomalies

5. Decrease in reactor coolant system inventory

6. Loss of coolant events

An increase in CEA rod drop time by 0.4 seconds has an impact primarily on those events which (a) involve a rapid approach to a safety limit during the same time frame as the scram, and/or (b) the event involves a rapid approach to a specified acceptable fuel design limit (minimum DNBR) during the first part of the scram insertion.

The limiting event for the increase in heat removed by the secondary system category (Category 1) is the inside containment steam line break pre-trip power excursion. This event was reanalyzed by the licensee. In addition to the 0.4 second increase in rod drop time assumed by the licensee, a 3 second delay in loss of AC power was assumed. Attachment 4 of the licensee's submittal provided a discussion of the reanalysis with assumptions used and final results. The difference in minimum DNBR was 0.001 (0.782 for "previous" analysis versus 0.783 for "new" analysis). The licensee concluded that predicted fuel failure is less than 10% and a coolable geometry is maintained. The licensee also evaluated the effect of only changing the rod drop time and determined that there would be a 3% degradation in minimum DNBR. The licensee concluded that the site boundary doses for this event are bounded by the doses obtained in the outside containment steam line break event. The site boundary doses for the outside steam line break event are well within the guidelines contained in 10 CFR Part 100. The staff has reviewed the licensee's analysis for the category of events and licensee's conclusions appear reasonable. There is a slight decrease in minimum DNBR but the acceptance criteria continues to be met. In addition, resultant doses are within 10 CFR Part 100. Therefore, the proposed change is acceptable considering the Category 1 events.

The limiting event for the decrease in heat removal by the secondary system category (Category 2) is the loss of condenser vacuum. This event was reanalyzed by the licensee. Attachment 4 of the licensee's submittal provided a discussion of the reanalysis with assumptions used and final results. The licensee concluded that the reactor coolant system (RCS) pressure and secondary system pressure does not exceed the acceptance criteria of 2750 psia and 1100 psia, respectively. The increase in RCS pressure was 18 psia and the increase in secondary system pressure was 2 psia. The staff has reviewed the licensee's analysis for this category of events and the licensee's conclusions appear reasonable. Since the pressure acceptance criteria for the primary and secondary systems continues to be met, the proposed change is acceptable considering the Category 2 event.

The licensee reanalyzed two decreases in reactor coolant flowrate events (Category 3): loss of forced reactor coolant flow (four reactor coolant pumps (RCP) tripped) and single sheared shaft (one RCP). Attachment 4 of the licensee's submittal provided a detailed discussion of the reanalysis of the loss of forced flow event with assumptions used and final results. A 3.5% decrease in overpower margin was calculated. The licensee concluded that since there is at least 5% overpower margin between actual calculated DNB LCO and the TS LCO, the 3.5% reduction can be accommodated without changing the existing DNB LCO in the TS. The licensee's conclusions appear reasonable and are acceptable. In regard to the sheared shaft event, the licensee determined that a 0.3% decrease in DNBR would occur but the DNBR would still be above the minimum DNBR. The licensee's results appear reasonable and are acceptable. Thus, the TS change in so far as Category 3 events are concerned is acceptable.

The limiting events for the reactivity and power distribution anomalies category (Category 4) is the CEA ejection event from hot full power. Attachment 4 of the licensee's submittal provided a detailed discussion of this event with assumptions used and final results. In addition to assuming an additional 0.4 seconds in rod drop times, the licensee made two other assumption changes beyond the analysis of record: reduce the post-ejected radial peaking factor from 3.5 to 3.2 to reflect a value more characteristic of the actual calculated values for recent cycles, and reduce the scram worth from $-4.5\%\Delta\rho$ to $-3.0\%\Delta\rho$ to accommodate expected reductions in available scram worth for future cycles. The licensee concluded that the peak average and centerline enthalpy calculated for the hottest pellet were both below all the fuel deposited energy limits, and no fuel failure was predicted. The licensee's conclusions are reasonable and are acceptable.

The licensee reviewed two decreases in RCS inventory events (Category 5): pressurizer pressure decrease and steam generator tube rupture. The licensee determined that the analysis of record for the pressurizer pressure decrease event still bounds the assumed increase in CEA rod drop time. The licensee determined that the increase in the CEDM holding coil delay has negligible impact on the calculated doses for the steam generator tube rupture event. These conclusions appear reasonable and the TS changes are acceptable as far as Category 5 events are concerned.

The licensee stated that the large break loss of coolant accident (LOCA) is not impacted by the increase in CEDM holding coil delay time and the small break LOCA is not impacted by the proposed amendment. The staff agrees with the licensee's statements for Category 6 events and the TS change is acceptable on this basis.

SUMMARY

The staff finds the proposed increase in CEA drop time acceptable based on the above evaluation of its effect on the referenced safety analyses. These analyses either remain bounding or continue to satisfy the staff's acceptance criteria. Therefore, the proposed Technical Specification change is acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the 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 impact 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: December 19, 1988

Principal Contributor:
E. Tourigny