DEC 2 1 1977

Docket No. 50-335✓

Florida Power & Light Company
ATTN: Dr. Robert E. Uhrig
Vice President
Nuclear & General Engineering
P. O. Box 013100
Miami, Florida 33101

Gentlemen:

The Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. DPR-67 for St. Lucie Unit No. 1. The amendment consists of changes to the Technical Specifications appended to License No. DPR-67 and changes to license conditions in Enclosure 1 in response to your applications referred to in the following paragraph.

The amendment:

1. Modifies the Offsite Organization Chart to accommodate changes which are administrative in nature in accordance with your application dated September 6, 1977 (L-77-277),

2. Reduces the Control Element Assembly specified drop time from 3.3 seconds to 3.0 seconds for consistency with times used to establish trip setpoints in accordance with your application dated June 30, 1977 (L-77-197) as supplemented by letter dated August 4, 1977 (L-77-247),

3. Modifies a transition limit to extend the transition period from 15 minutes to one hour without either a reactor coolant pump or a shutdown cooling pump running in accordance with your application dated September 30, 1977 (L-77-309),

4. Deletes license condition F.1 (which required installation of gates/valves to control water flow in the ultimate heat sink barrier dam) in accordance with your application dated July 18, 1977 (L-77-225), and

5. Deletes license condition G.1 (which required installation of erosion protection for part of the discharge canal) in accordance with your application dated October 25, 1977 (L-77-331).

Const 1
GD

OFFICE➤						
SURNAME➤						
DATE➤						

Our review of your applications resulted in minor modifications to your proposed changes which have been discussed with and agreed to by your staff.

Copies of the related Safety Evaluation and our Notice of Issuance of the amendment also are enclosed.

Sincerely,

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 18 to DPR-67
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

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660 Apalachee Parkway
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Mr. Hamilton Oven, Jr., Administrator -- w/cy of FPL's filings referenced on pg. 1 of this letter.
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U. S. Environmental Protection Agency
Region IV Office
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345 Courtland Street, N. E.
Atlanta, Georgia 30308

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
License No. DPR-67

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by the Florida Power & Light Company (the licensee) dated June 30, 1977 (as supplemented by letter dated August 4), July 18, September 6 and 30, and October 25, 1977, comply 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 applications, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by the following changes to Facility Operating License No. DPR-67:

   A. Change paragraph 2.C(2) in its entirety to read as follows:

      (2) <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 18, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   B. Delete in their entirety conditions F and G of Enclosure 1 appended to the license.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: December 21, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 18

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.

Pages

3/4 1-26
3/4 4-1
B 3/4 1-4
6-2
6-9

POSITION INDICATOR CHANNELS (Continued)

LIMITING CONDITION FOR OPERATION

b) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.

c. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel at either its fully inserted position or fully withdrawn position, operation may continue provided:

1. The position of this CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable),

2. The fully inserted CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted, and

3. Subsequent operation is within the limits of Specification 3.1.3.6.

d. With one or more pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 4.5 inches at least once per 12 hours except during time intervals when the Deviation circuit is inoperable, then compare the pulse counting position indicator and reed switch position indicator channels at least once per 4 hours.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.0 seconds from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, 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 of the reactor vessel head,

b. For specifically affected individual 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.

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Four reactor coolant pumps shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

MODES 1 and 2:

With less than four reactor coolant pumps in operation, be in at least HOT STANDBY within 6 hours.

MODES 3, 4 and 5:

Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump.# The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1 The Flow Dependent Selector Switch shall be determined to be in the 4 pump position within 15 minutes prior to making the reactor critical and at least once per 12 hours thereafter.

---

* See Special Test Exception 3.10.4.

\# All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA $\pm$ 1%.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

3/4.1.2 BORATION SYSTEMS (Continued)

The boron capability required below 200°F is based upon providing a 1% Δk/k SHUTDOWN MARGIN at 140°F during refueling with all full and part length control rods withdrawn. This condition requires either 5,650 gallons of 8.0% boric acid solution from the boric acid tanks or 100,000 gallons of 1720 ppm borated water from the refueling water tank.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment (≥ 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (≥ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time permitted by Specification 3.1.3.4 is the assumed CEA drop time of 3.0 seconds used in the safety analyses. Measurement with $T_{avg} \geq 515°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

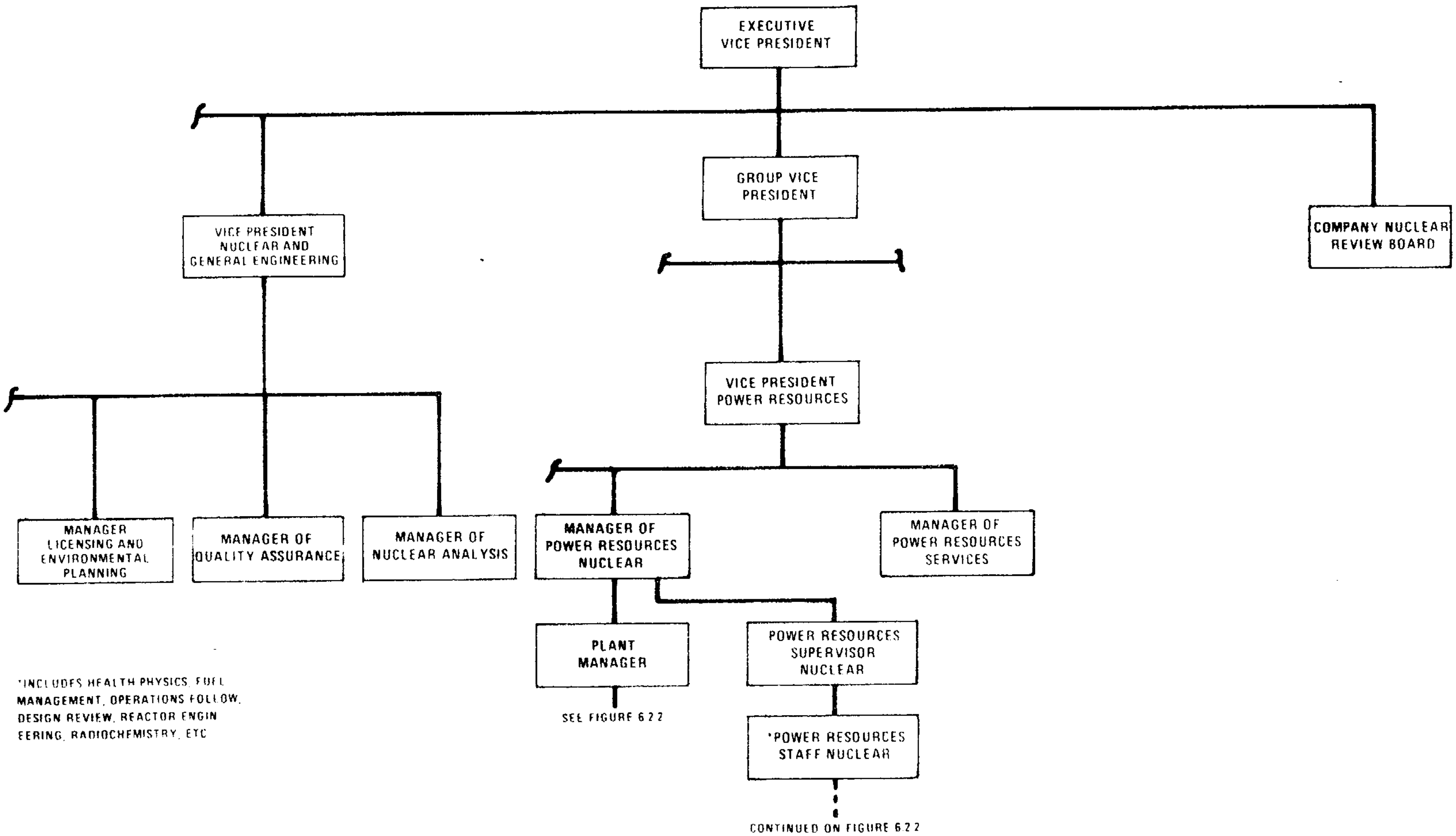

Figure 6.2-1 Offsite Organization for Facility Management and Technical Support

COMPOSITION

6.5.2.2 The CNRB shall be composed of the:

Member:	Vice President Nuclear and General Engineering
Member:	Chief Engineer Power Plants
Member:	Vice President of Power Resources
Member:	Power Plant Engineering Supervisor
Member:	Manager of Power Resources - Nuclear
Member:	Manager of QA
Member:	Power Plant Engineering Supervisor

The CNRB Chairman shall be designated in writing.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of CNRB shall consist of the Chairman or his designated alternate and four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The CNRB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. REPORTABLE OCCURRENCES requiring 24 hour notification to the Commission.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meetings minutes of the Facility Review Group.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT NO. 1

DOCKET NO. 50-335

INTRODUCTION

By the following applications, Florida Power and Light Company (FPL) requested amendments to St. Lucie Unit No. 1 License No. DPR-67. The amendments would change the Technical Specifications to:

1. Modify the Offsite Organization Chart to accommodate changes which are administrative in nature in accordance with your application dated September 6, 1977 (L-77-277),

2. Reduce the Control Element Assembly specified drop time from 3.3 seconds to 3.0 seconds for consistency with times used to establish trip setpoints in accordance with your application dated June 30, 1977 (L-77-197) as supplemented by letter dated August 4, 1977 (L-77-247),

3. Modify a transition limit to extend the transition period from 15 minutes to one hour without either a reactor coolant pump or a shutdown cooling pump running in accordance with your application dated September 30, 1977 (L-77-309),

4. Delete license condition F.1 (which required installation of gates/valves to control water flow in the ultimate heat sink barrier dam) in accordance with your application dated dated July 18, 1977 (L-77-225), and

5. Delete license condition G.1 (which required installation of erosion protection for part of the discharge canal) in accordance with your application dated October 25, 1977 (L-77-331).

Our review of the applications resulted in minor modifications to FPL's proposed changes. These changes have been discussed with the FPL staff who agreed with the modifications.

DISCUSSION AND EVALUATION

1. Offsite Organizational Changes

   By letter dated September 6, 1977, FPL proposed Technical Specifications changes to the Offsite Organization Chart. Proposed changes to figure 6.2-1 are as follows:

   a. New title -- Vice President Nuclear and General Engineering

   b. New title -- Manager Licensing and Environmental Planning

   c. Position deleted -- Vice President Environmental Planning and Research

   FPL states that the changes are not significant changes in the management or support of the FPL nuclear facilities; the responsibilities for research and environmental planning have been transferred to different people; and the changes have been reviewed by the St. Lucie Facility Review Group and the FPL Nuclear Review Board; and that the changes do not involve an unreviewed safety question.

   Since the changes would not degrade operational plant safety, are administrative in nature, and would not affect the operation of either on-site or off-site safety review functions, the changes are acceptable.

2. Control Element Assembly (CEA) Drop Time

   The individual full length (shutdown and control) CEA drop time, from fully withdrawn to 90 percent inserted, is a limiting condition of operation. A surveillance requirement is accomplished following reactor head removal, when maintenance or modifications are accomplished, and at least once per 18 months. This surveillance assures operability of the CEA's to reduce reactivity quickly when a Reactor Protection System signal requires a reactor trip. FPL proposed by letter dated June 30, 1977, to change the specified time from 3.3 seconds to 3.0 seconds in Specification 3.1.3.4.

The proposed change is consistent with the assumed CEA drop time Combustion Engineering used for the analysis of the most limiting Anticipated Operational Occurrence. The 3.0 seconds drop time was used to establish the trip setpoints for St. Lucie Unit No. 1. However, the current revision of the Final Safety Analysis Report for St. Lucie indicates 3.3 seconds as the assumed CEA drop time in the accident analyses. FPL letter of August 4, 1977 reports that about 2.4 seconds was the slowest time observed during startup tests at St. Lucie. Thus, since (1) certain trip setpoints were based on 3.0 seconds CEA drop time, (2) startup tests confirm the 3.0 seconds is achievable, and (3) a faster CEA drop time (3.0 vice 3.3 seconds) results in more conservative safety considerations, the change is acceptable.

3. <u>Reactor Coolant Pump or Shutdown Cooling Pump Operation</u>

By letter dated September 30, 1977, FPL proposed Technical Specification changes to allow operation in Modes 3, 4 and 5 (hot standby, hot shutdown or cold shutdown) without either a coolant pump or a shutdown cooling pump running for a period dependent upon core temperature and pressure. The existing Specification 3.4.1 allows 15 minutes of operation (transition period) without either a reactor coolant pump or a shutdown cooling pump running.

Our evaluation indicates that the existing 15-minute transition period is too restrictive for normal operation during heatup and cooldown transitions. We agree with FPL's desire to increase the transition time period. However, we do not believe that the proposed deletion of a specific transition time limit, based on added monitoring of the core decay heat parameters during the transient period would be productive or in the interests of maintaining the existing level of plant safety. The operator would be required to evaluate additional information and assess the safety impact during the period when neither a coolant pump nor a shutdown cooling pump is running. Such added operator action is considered unnecessary.

We consider that an increase from 15 minutes to one hour would provide an adequate and safe transition period. Such an increase would be conditioned by requirements which we would impose to assure that no action is taken which could result in boron dilution during the transition period.

FPL stated that natural circulation would provide thorough mixing of borated water within the reactor coolant system when pumps are secured. However, test data to substantiate the degree of mixing at St. Lucie was not provided. Such data is not needed because of the added precautionary condition to require that no operations be permitted which could cause dilution of the system boron concentration. Therefore, for the reasons cited above and because the level of safety has not been decreased, we conclude that the proposed change to Technical Specification 3.4.1, as modified, is acceptable. FPL personnel have agreed to the modifications.

4. Completion of License Conditions

Enclosure 1 to License No. DPR-67 for St. Lucie Unit No. 1 identified certain items to be completed to the satisfaction of the Commission. The following conditions have been satisfied and are deleted from License No. DPR-67:

(1) Condition F.1 required installation of the gates/valves to control water flow in the ultimate heat sink (UHS) barrier dam by March 31, 1977. On March 25, 1977 we issued Amendment No. 14 to FPL's license extending until July 31, 1977 the March 31, 1977 deadline for completion of the installation. The safety evaluation issued with Amendment No. 14 concluded that the 4-month extension would have a negligible effect on plant safety.

By letter dated July 18, 1977, FPL advised us that flow control valves had been installed in the UHS dam. Based on the completion of installation of the valves and on our previous acceptability of the design, we conclude that license Condition F.1 has been met and may be deleted from Enclosure 1 to the license.

(2) Condition G.1 required installation of erosion protection for part of the discharge canal by June 30, 1977. On May 24, 1977, we responded to FPL's proposal of March 18, 1977, to redesign the erosion protection to delete the concrete pile cap. Our conclusion was that the redesign was not acceptable since the concrete cap was an essential part of the design. We approved extending until Octover 31, 1977 the concrete cap installation.

By letter dated October 25, 1977 FPL advised us that the required erosion protection had been installed. Based on the completion of the installation of the previously approved design of erosion protection for part of the discharge canal peninsula, we conclude that license Condition G.1 has been met and may be deleted from Enclosure 1 to the license.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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 21, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-335

FLORIDA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 to Facility Operating License No. DPR-67, issued to Florida Power & Light Company (the licensee), which revised the Technical Specifications for operation of the St. Lucie Plant Unit No. 1 (the facility) located in St. Lucie County, Florida. The amendment is effective as of its date of issuance.

The amendment revised the Technical Specifications to: (a) modify the Offsite Organizational Structure, (b) reduce from 3.3 seconds to 3.0 seconds the Control Element Assembly drop time, (c) modify the transition limit to extend the transition period from 15 minutes to one hour without either a reactor coolant pump or shutdown cooling pump running, and (d) delete conditions F and G of Enclosure 1 to the license since these items (relating to control of water flow in the ultimate heat sink and installation of erosion protection for part of the discharge canal) have been satisfactorily completed.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated June 30, 1977 (as supplemented by letter dated August 4); July 18, September 6 and 30, and October 25, 1977, (2) Amendment No. 18 to License No. DPR-67, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida 33450. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 21th day of December, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors