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Docket Nos. 50-250
and 50-251

Dr. Robert E. Uhrig, Vice President
Advanced Systems and Technology
Florida Power and Light Company
Post Office Box 529100
Miami, Florida 33152

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. DPR-31 and Amendment No. 51 to Facility Operating License No. DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letters dated March 14 (L-80-83), and June 5, 1980 (L-80-170 and L-80-171).

These amendments incorporate the results of a revised ECCS analysis for a steam generator plugging level of 25% for both Units 3 and 4 and permit continued operation of Unit 4 for six equivalent months of operation from June 11, 1980, at which time the steam generators for Unit 4 shall be inspected.

In view of the fact that computational errors have occurred three times in less than a year, we request that you inform us of your plans to prevent recurrence of computational errors in the future.

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

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 58 to DPR-31
2. Amendment No. 51 to DPR-41
3. Safety Evaluation
4. Notice of Issuance

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* SEE PREVIOUS YELLOW ~~FOR~~ FOR CONCURRENCE

OFFICE	cc w/enclosures See next page	DL:ORB1*	DL:ORB1*	DL:ORB1	OELD*	DL:AD:OR*
SURNAME		MGrotenhuis:jb	CSParrish	SAVarga	SGoldberg	TMNovak
DATE		06/06/80	06/2 /80	06/09/80	06/09/80	06/09/80

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

June 12, 1980

Docket Nos. 50-250
and 50-251

Dr. Robert E. Uhrig, Vice President
Advanced Systems and Technology
Florida Power and Light Company
Post Office Box 529100
Miami, Florida 33152

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Division of Licensing

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2. Amendment No. 51 to DPR-41
3. Safety Evaluation
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cc: w/enclosures
See next page

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County Manager of Metropolitan Dade County
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 58
License No. DPR-31

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated June 5, 1980 (L-80-171), 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-31 is hereby amended to read as follows:

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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 58 , 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

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1980

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. DPR-31

DOCKET NO. 50-250

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.2-3	3.2-3
3.2-4	3.2-4
Figure 3.2-3	Figure 3.2-3
3.2-3a	
3.2-3b	

reactivity insertion upon ejection greater than 0.3% k/k at rated power. Inoperable rod worth shall be determined within 4 weeks.

b. A control rod shall be considered inoperable if
   (a) the rod cannot be moved by the CRDM, or
   (b) the rod is misaligned from its bank by more than 15 inches, or
   (c) the rod drop time is not met.

c. If a control rod cannot be moved by the drive mechanism, shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the inoperable rod.

5. CONTROL ROD POSITION INDICATION

If either the power range channel deviation alarm or the rod deviation monitor alarm are not operable, rod positions shall be logged once per shift and after a load change greater than 10% of rated power. If both alarms are inoperable for two hours or more, the nuclear overpower trip shall be reset to 93% of rated power.

6. POWER DISTRIBUTION LIMITS

a. Hot channel factors:

With steam generator tube plugging $\leq 25\%$, the hot channel factors (defined in the basis) must meet the following limits at all times except during low power physics tests:

$$F_q (Z) \leq (1.93/P) \times K(Z), \text{ for } P > .5$$

$$F_q (Z) \leq (3.86) \times K(Z), \text{ for } P \leq .5$$

$$F^N_H \leq 1.55 [1.+0.2 (1-P)]$$

Where P is the fraction of rated power at which the core is operating; K(Z) is the function given in Figure 3.2-3; Z is the core height location of F_q.

If F_q, as predicted by approved physics calculations, exceeds 1.93, the power will be limited to the rated power multiplied by the ratio of 1.93 divided by the predicted F_q, or augmented surveillance of hot channel factors shall be implemented.

b. Following initial loading before the reactor is operated above 75% of rated power and at regular effective full rated power monthly intervals thereafter, power distribution maps, using the movable detector system shall be made, to confirm that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

(1) The measurement of total peaking factor, F_q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

(2) The measurement of the enthalpy rise hot channel factor, F^N_H, shall be increased by four percent to account for measurement error.

If either measured hot channel factor exceeds its limit specified under Item 6a, the reactor power shall be reduced so as not to exceed a fraction of the rated value equal to the ratio of the F_q or F^N_H limit to measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio. If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing. The reactor may be returned to higher power levels when measurements indicate that hot channel factors are within limits.

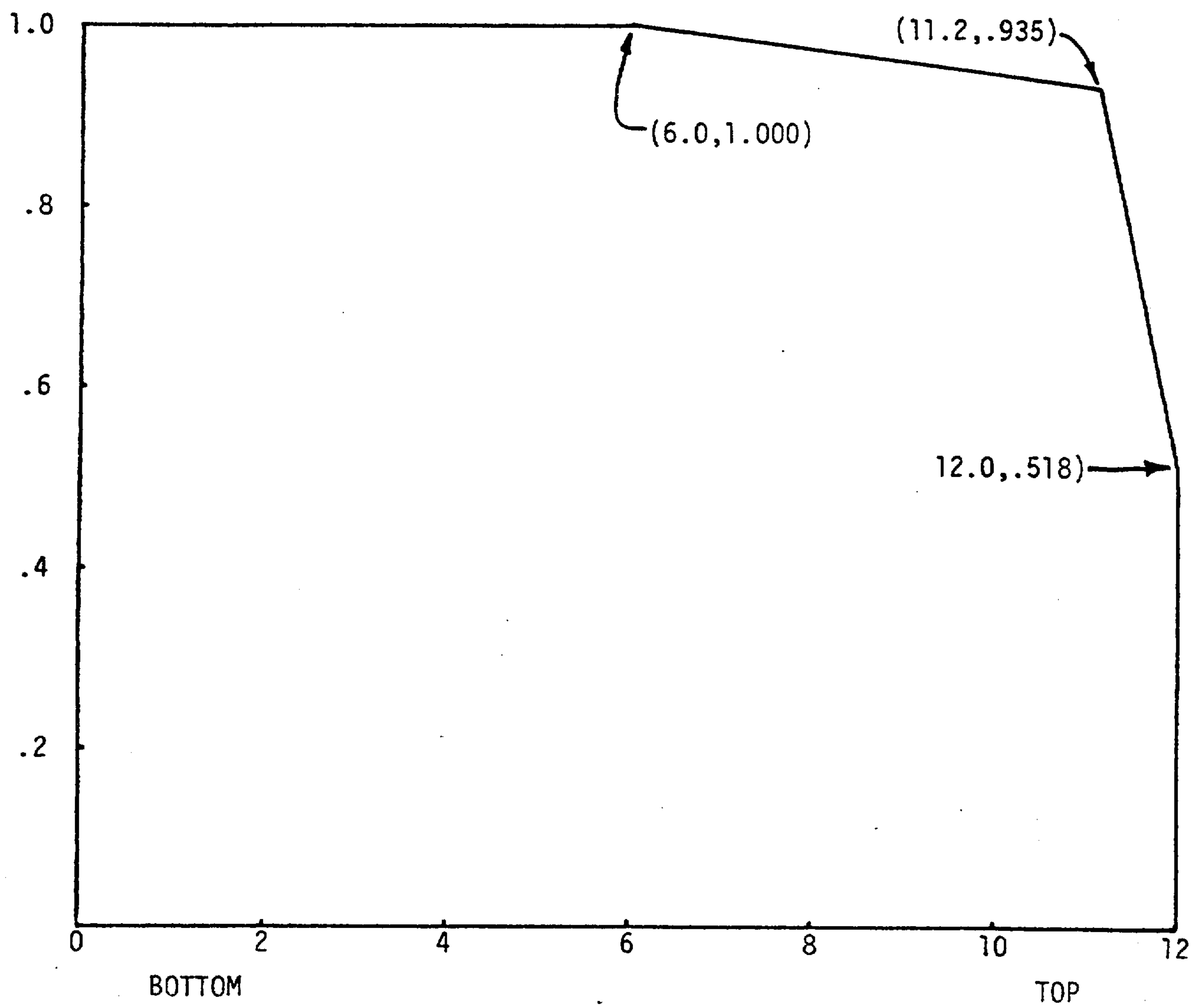
c. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at lest once per effective full power quarter. If the axial flux difference has not been measured in the last effective full power month, the target flux difference must be updated monthly by linear interpolation using the most recent measured value and the value predicted for the end of the cycle life.

d. Except during physics tests or during excore calibration procedures and as modified by items 6e through 6g below, the indicated axial flux differene shall be maintained within a $\pm$ 5% band about the target flux difference (this defines the target band on axial flux difference).

e. If the indicated axial flux difference at a power level greater than 90% of rated power deviates.

HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

(for $\leq$25% steam generator tube plugging and F_q=1.93)



CORE HEIGHT (FT.)

Figure 3.2-3

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. DPR-41

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

   A. The applications for amendment by Florida Power and Light Company (the licensee) dated March 14 (L-80-83), and June 5, 1980 (L-80-170 and L-80-171), 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 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 3.B, 3.D.1 and 3.D.2 of Facility Operating License No. DPR-41 are hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 51 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

D. Steam Generator Operation

(1) After operation in Cycle 6 of six equivalent full power months from June 11, 1980, Turkey Point Unit 4 shall be brought to the cold shutdown condition and the steam generators shall be inspected unless: (1) an inspection of the steam generators is performed within this period as a result of the requirements in 2, 3 and 4 below, or (2) an acceptable analysis of the susceptibility for stress corrosion cracking of tubing is submitted to explicitly justify continued operation of Unit No. 4 beyond the authorized period of operation. Any analysis justifying continued operation must be submitted at least 45 days prior to the expiration date of the authorized period of operation. For the purpose of this requirement, equivalent operation is defined as operation with the reactor coolant at a temperature greater than 350°F. Nuclear Regulatory Commission (NRC) approval shall be obtained before resuming power operation following this inspection.

(2) Reactor coolant to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm per steam generator. With a steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours. A full steam generator inspection shall be performed and NRC approval shall be obtained before resuming power operation following this inspection.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1980

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-251

Replace the following pages of the Facility Operating License No. DPR-41 with the attached pages as indicated. The changed area in the license is indicated by a marginal line.

Remove Pages	Insert Pages
4	4
5	5

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.2-3	3.2-3
3.2-4	3.2-4
Figure 3.2-3	Figure 3.2-3
3.2-3a	
3.2-3b	

B. Technical Specifications

The Technical Specifications contained in Appendices A and B as revised through Amendment No. 43 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following conditions for the protection of the environment:

(1) The applicant shall pursue evaluations of alternatives to the proposed cooling channel system during construction, interim operation, and evaluation of the channel system. These evaluations shall include at least the following:

(a) Study of availability of groundwater or other alternative sources of surface water to use in the cooling system.

(b) Study of applicability of mechanical cooling devices, including powered spray modules and cooling towers.

(c) Study of marine environmental impacts of once-through cooling alternatives (described in Section X of the AEC Final Environmental Statement on Turkey Point Units 3 and 4, July 1972).

(2) The applicant shall take appropriate corrective action on any adverse effects determined as a result of monitoring and study programs. To the fullest extent practicable, the applicant shall utilize results of study programs in improving and modifying the operation of the facility and its cooling system so as to achieve a minimal adverse environmental impact.

D. Steam Generator Operation

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(3) The concentration of radioiodine in the reactor coolant shall be limited to 1.0 microcurie/gram during normal operation and to 30 microcuries/gram during power transients.

(4) Reactor operation shall be terminated and NRC approval shall be obtained prior to resuming operation if primary to secondary leakage attributable to the denting phenomena is detected in 2 or more tubes during any 20 day period.

(5) The Metal Impact Monitoring System (MIMS) shall be contained in operation with the capability of detecting losse objects. If the MIMS is out of service in other than cold shutdown or refueling mode of operation, this fact shall be reported to the NRC. Any abnormal indications from the MIMS shall also be reported to the NRC by telephone by the next working day and by a written evaluation within two weeks.

(6) Following each startup from below 350°F, core barrel movement shall be evaluated using neutron noise techniques.

reactivity insertion upon ejection greater than 0.3% k/k at rated power. Inoperable rod worth shall be determined within 4 weeks.

b. A control rod shall be considered inoperable if
   (a) the rod cannot be moved by the CRDM, or
   (b) the rod is misaligned from its bank by more than 15 inches, or
   (c) the rod drop time is not met.

c. If a control rod cannot be moved by the drive mechanism, shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the inoperable rod.

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Where P is the fraction of rated power at which the core is operating; K(Z) is the function given in Figure 3.2-3; Z is the core height location of F_q.

If F_q, as predicted by approved physics calculations, exceeds 1.93, the power will be limited to the rated power multiplied by the ratio of 1.93 divided by the predicted F_q, or augmented surveillance of hot channel factors shall be implemented.

b. Following initial loading before the reactor is operated above 75% of rated power and at regular effective full rated power monthly intervals thereafter, power distribution maps, using the movable detector system shall be made, to confirm that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

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If either measured hot channel factor exceeds its limit specified under Item 6a, the reactor power shall be reduced so as not to exceed a fraction of the rated value equal to the ratio of the F_q or F^N_H limit to measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio. If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing. The reactor may be returned to higher power levels when measurements indicate that hot channel factors are within limits.

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e. If the indicated axial flux difference at a power level greater than 90% of rated power deviates.

HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

(for $\leq$25% steam generator tube plugging and F_q=1.93)

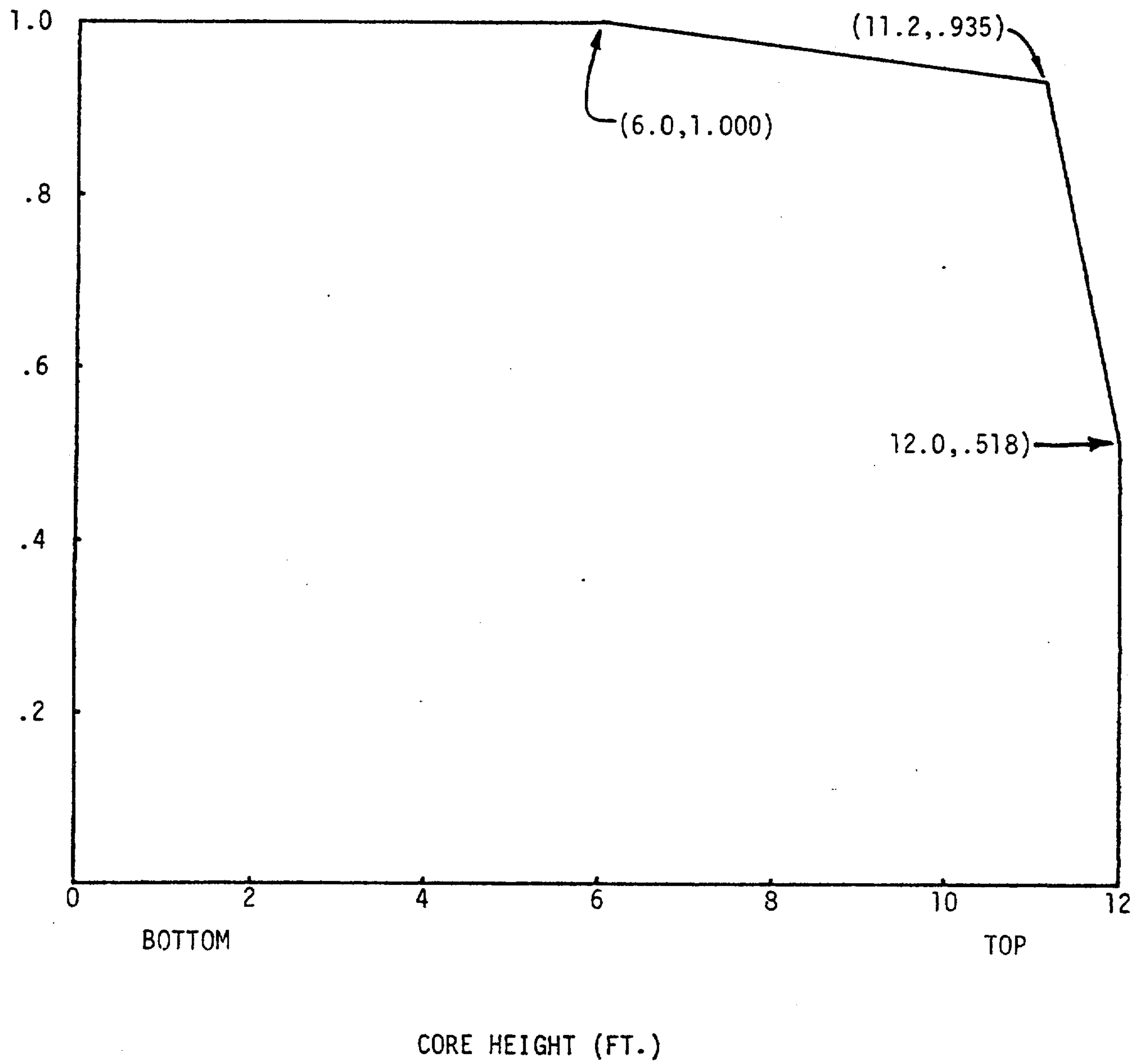

Figure 3.2-3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT PLANT UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

Introduction

By letter dated March 14, 1980 (L-80-83) Florida Power and Light Company (the licensee) submitted the steam generator inspection program for the Turkey Point Plant Unit No. 4. On June 5, 1980 (L-80-170) the licensee submitted the results of the most recent steam generator inspection and requested authorization to operate Unit 4 for six equivalent full power months*, beginning on or about June 12, 1980, at which time a steam generator inspection will be performed.

The technical basis for the preventive maintenance plugging program implemented subsequent to the inspection was consistent with that for programs performed previously at this and other similarly degraded units. These programs have been determined adequate by the NRC to support six (6) equivalent months of operation.

By letter (L-80-171) dated June 5, 1980 (reference 1), the licensee also requested amendments to Operating Licenses DPR-31 and DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4. This letter also contained a LOCA analysis and proposed Technical Specifications changes in connection with operation of Units 3 and 4 with 25% of the steam generator tubes plugged and a peaking factor, F_Q, of 1.93.

Part I
Unit 4 Steam Generator Inspection

Discussion

The steam generator tube inspection performed at Turkey Point Unit 4 during May 1980 included programs to assess tube degradation associated with both the denting and wastage phenomena. For denting, tube gauging was performed in all

*Equivalent operation is defined as operation with the reactor coolant greater than 350°F.

The licensee submittals refer to "effective full power" months which we interpret to mean "equivalent months."

three steam generators using .650 inch, .610 inch, and .540 inch (diameter) eddy current probes. The implemented gauging program was similar to those implemented previously at this and other similarly degraded units and included the gauging of all unplugged tubes within areas (tubelane, periphery, wedge, and patch plate regions of the hot leg, and tubelane region of the cold leg) where significant denting activity had been observed previously. Significant denting, in this context, is considered to include (in addition to leakers) tubes restricting passage of a .610 inch probe (or less) and tubes at the periphery of the hot leg wedge location and on either side of the patch plate boundary which restrict passage of a .650 inch probe (or less), since these tubes are the most likely candidates to develop inservice leaks.

In previous inspections of the tubelane region, finite element analysis had been used to determine the extent of significant tube restriction activity for purposes of defining the boundary for the tubelane gauging inspection. However, the 17.5% tube hoop strain contour which realistically bounded the significant tube restriction activity in the tubelane following the previous inspection is now predicted to cover most of the support plate. Thus, the licensee elected to gauge the tubelane tubes within a boundary incorporating previously observed activity, plus several rows of tubes beyond.

With regards to the defined regions (discussed above) within which all tubes were gauged, if a restricted tube (tube restricting a .650 inch probe) was found close to the inspection boundary, the inspection was expanded in that area. In addition, a sample population of tubes in the central bundle region, located outside these defined regions, was tested with .700 and .720 inch probes in the hot and cold legs, respectively, as part of the Regulatory Guide 1.83 eddy current inspection (to be discussed). These latter inspections provide an early indication of any new deformation which exists away from the regions usually regarded as active (i.e. the tubelane, patch plate, wedges, and periphery).

Measurements of the visible support plate flow slots in all steam generators were made to assess the condition of the support plates and to provide a gross measure of the continuation of denting.

Eddy current inspection for wastage was conducted in accordance with Regulatory Guide 1.83 in all of the steam generators. Eddy current examinations were also performed on the U-bends of the unplugged tubes in rows three through five in steam generator B.

The following tabulation summarizes the number of tubes included in the gauging and eddy current inspections:

	A Hot Leg	A Cold Leg	B Hot Leg	B Cold Leg	C Hot Leg	C Cold Leg
Gauging	1205	170	1276	140	1209	179
U-band Rows 2-5	-	-	-	55	-	-
R.G. 1.83	301	360	283	164	146	275

INSPECTION RESULTS

The results of the gauging inspection in terms of the number of tubes restricting passage of a given size probe of .650 inch or less are summarized below:

	Tubelane		Periphery and Wedge		Patch Plate
	Hot leg	Cold leg	Hot leg	Cold leg	Hot leg
SG A					
.650"	20	0	22	0	2
.610"	15	0	5	0	0
.540"	4	0	2	0	0
SG B					
.650"	23	0	32	0	6
.610"	9	0	5	0	0
.540"	3	0	2	0	0
SG C					
.650"	28	0	23	0	12
.610"	14	4	5	2	4
.540"	3	0	1	0	0

Tubes in the tubelane region that restrict a .650 inch probe or less are located adjacent to the areas in which such restrictions have been observed during previous inspections. Tube restriction activity observed in the periphery, wedge, and patch plate areas appears consistant with previous experience at this and other similarly degraded units. The level of denting activity in the cold leg remains low compared to the hot leg experience.

In steam generator B, four tubes restricting passage of a .650 probe or less are located away from the pattern of prior results. These restrictions were identified during the Regulatory Guide 1.83 eddy current inspection program. The gauging inspection was subsequently expanded to include neighboring tubes. Three of these restrictions occurred adjacent to a tube which had been pulled in September 1975 as part of a field investigation of denting. The forth tube was located near the patch plate region.

The Regulatory Guide 1.83 inspection also identified a number of tubes restricting passage of either a .720 or .700 inch probe as follows:

	Hot Leg	Cold Leg
SG A	4	6
SG B	41	0
SG C	11	0

Except for the four tubes discussed above, the affected hot leg tubes successfully passed a .650 inch probe. The affected cold leg tubes were not regauged with a .650 inch probe, but did successfully pass a .610 inch probe.

No forced shutdowns because of tube leakage occurred during the ten EFPM of operation since the previous inspection in April 1979. However, a potential leak indication of 0.032 gal/hr was reported on March 20, 1980. This is below the threshold for detection which is considered to be 0.10 gal/hr. The leakage limit permitted by the license is 0.3 gal/min or 1.8 gal/hr. A hydrostatic leak test indicated a peripheral tube in steam generator A, which had previously passed a .650 inch probe, as the probable source of the leakage. This tube and the surrounding tubes were plugged.

The random eddy current inspection performed in accordance with Regulatory Guide 1.83 identified 14 tubes that required plugging due to thinning indications. The licensee attributes the finding of the 14 pluggable indications primarily on the differences between eddy current data interpreters employed during the previous inspection versus those employed during this inspection, rather than on significant degradation of the tubes since the previous inspection. This conclusion is based upon a reevaluation of the previous April 1979 eddy current data from which it was determined that the average difference in wall thinning between the April 1979 and the May 1980 inspection was only approximately 1%.

No eddy current indications were identified in the U-bends of the unplugged tubes in Rows 3 through 5.

The measurements of the support plate flow slots indicated no deviations from anticipated conditions.

TUBE PLUGGING PROGRAM

Except as noted below, the plugging criteria implemented during the May 1980 steam generator inspection are the same as those implemented previously at this and other similarly degraded units to support six months operation. These criteria include the plugging of leakers and surrounding tubes, .540 inch and .610 inch restricted tubes, .650 inch restricted tubes in the periphery of the hot leg wedge region and on either side of the patch plate boundary.

Previous criteria for the tubelane region (to support six months of operation) had included plugging two tubes beyond .540 inch restricted tubes in columns 15 through 79, and three tubes beyond .540 inch restricted tubes in columns 1 to 14 and 80 to 94, based upon finite element predictions regarding the progression of denting during the next operating interval. The licensee stated in a letter dated May 18, 1979 that wedge and tubelane interaction was apparently causing the finite element analysis to over predict the progression of denting in the end regions of the tubelane based upon their evaluation that denting activity in these end regions are consistent with the remainder of the tubelane region. The implemented criterion for this inspection was to plug two rows beyond .540 inch restricted tubes for columns 1 through 92. The licensee has stated (in their June 8, 1980 letter) that their review of the data from all previous inspections indicates that this rate of progression (two rows per six months for .540 inch restricted tubes) is not occurring on a general basis.

Finally, tubes with greater than 40% through wall eddy current indications were plugged.

Implementation of the plugging criteria resulted in 64, 45, and 53 tubes being plugged for denting and 2, 0, and 2 tubes being plugged for wastage in steam generators A, B, and C, respectively. Total steam generator tube plugging in all three steam generators is approximately 22.4% which is conservatively bounded by the 25% tube plugging ECCS analysis.

EVALUATION

The May 1980 gauging and preventive plugging program at Turkey Point Unit 4 is similar to previous programs conducted in at this and other similarly degraded units. This inspection included the gauging of all tubes within areas (tubelane, periphery, wedge, and patch plate regions) where significant denting activity has been observed previously. In addition, a sample population of tubes in the central bundle region were gauged as part of the Regulatory Guide 1.83 inspection for wastage.

Based upon our review of the gauging results, we find that the observed denting activity is generally consistent with previous experience at this and other similarly degraded units, and that the implemented gauging program was sufficient to adequately determine the condition of the steam generators from a denting standpoint. Tube gauging performed as part of the Regulatory Guide 1.83 eddy current inspection did reveal a significant number of tubes in the central tube bundle region, particularly in the hot leg of steam generator B, which restricted passage of either a .720 inch or .700 inch probe. Of these, only four tubes in the hot leg of steam generator B restricted passage of a .650 inch probe or less and three of these four tubes appear to reflect a local phenomenon surrounding a previously removed tube. The gauging inspection was expanded in the vicinity of these four tubes and confirmed that these tubes were not indicative of a general condition of significant tube restriction activity in the central bundle region. However, the large number of .700 and .720 inch restricted tubes in the central bundle region is indicative of an early stage of denting in this region.

The preventive plugging criteria implemented in May 1980 and in previous inspections have proven successful in removing from service severely restricted tubes which are the most likely candidates to develop inservice leaks. The inspection data and recent operating experience provide adequate justification for the implemented criterion of plugging two tubes beyond .540 inch restricted tubes in the tubelane. We find that the implemented gauging program and preventive plugging criteria provide reasonable assurance that the vast majority of tubes most likely to develop inservice leaks have been identified and removed from service. Through wall cracks which have occurred at dented locations have been small and stable (no rapid failures). The license condition 0.3 gpm leak rate limit provides adequate assurance that even if through wall cracks and leaks occur, they will be detected and appropriate corrective action taken before any individual crack becomes sufficiently large as to be unstable under normal operating, transient, or accident conditions.

With regards to the wastage phenomenon, the May 1980 wastage inspection (per Regulatory Guide 1.83) and associated plugging criteria are similar to those implemented in previous inspections. We find the licensee has provided a satisfactory explanation of the 11 pluggable indications as being primarily a reflection of differences between eddy current data interpreters between inspections, rather than a reflection of significant recent degradation of the tubes as supported by the licensee's reevaluation of the previous eddy current data. We consider that the May 1980 inspection was adequate to establish the condition of the steam generators from a wastage standpoint and that with the implemented plugging criteria provides reasonable assurance that unacceptable wastage degradation will not occur during the next operating interval.

In conclusion, we find that the inspection results, implemented plugging, and existing leak rate limits adequately support six equivalent months of operation from the time of this inspection. We require that Turkey Point Unit 4 be required to shutdown for steam generator inspection at the conclusion of the six (6) month operating interval.

Part II
Steam Generator Tube Plugging Limit

Background

On November 9, 1979, the licensee was notified by Westinghouse (W), the NSSS vendor, that an input error had been identified in each of two LOCA analyses specifically applicable to the Turkey Point Plant Unit Nos. 3 and 4. The LOCA analysis for both the 22% and 25% steam generator tube plugging levels were affected. On November 23, 1979, LER-250-79-35 was issued stating that W had found that a non-conservative error could exist in the 10 CFR Part 50 Appendix K LOCA analysis with respect to the part of the calculation related to rod burst. Augmented surveillance was applied for both units according to the Technical Specifications and the F_Q was reduced appropriately by administrative means to compensate for the above errors.

On March 13, 1980, Amendment Nos. 54 and 46 were issued to Unit Nos. 3 and 4 respectively, which corrected the above errors for the 22% steam generator tube plugging limit. On May 15, 1980, Amendment Nos. 57 and 50 were issued to Unit Nos. 3 and 4 respectively, which corrected the above errors for the 25% steam generator tube plugging limit.

On May 28, 1980, the licensee notified NRR/DL/ORB1 that late the day before W notified them of an input error in the SATAN code that made the above recent amendments invalid. The error was corrected by a reduction in F_Q by 0.06, which was a conservatively chosen number based on a recalculation with proper input. The June 5, 1980 (L-80-171) amendment request was made to correct this error.

In view of the fact that computational errors have occurred three times in less than a year we will request that the licensee inform us of plans made to prevent the recurrence of computational errors.

The licensee has provided a LOCA analysis and proposed Technical Specification changes in connection with the operation of Units 3 and 4 with 25 percent of steam generator tubes plugged and a peaking factor F_Q of 1.93. In addition, the licensee provided sensitivity study indicating that the penalty caused by introducing the new fuel performance models developed by the NRC (Reference 2) is compensated by the conservatisms existing in the present ECCS models (Reference 1) and therefore no reduction of F_Q due to this effect is required.

The changes to the Technical Specifications requested by the licensee are the following:

(a) Specification of F_Q = 1.93 for plant operation with 25 percent of steam generator tubes plugged.

(b) Change of the Hot Channel Factor Normalized Operating Envelope for a steam generator tube plugging level of as many as 25 percent (Figure 3.2-3).

(c) Deletion of the specification for F_Q and the Hot Channel Factor Normalized Operating Envelope corresponding to a steam generator tube plugging level of 22 percent (Figure 3.2-3a).

Since the limiting value of F_Q is below the level at which the excore detectors could provide reliable readings and because the "18 case FAC analyses" performed for both units indicated that the maximum predicted F_Q exceeded the LOCA determined limits, the licensee is required either to operate the plant with the augmented power distribution surveillance or at the suitably reduced power levels.

Evaluation

The licensee has provided an evaluation of the performance of Emergency Core Cooling System (ECCS) for both Units 3 and 4 corresponding to the hot channel peaking factor value of F_Q = 1.93 and assuming a steam generator plugging level of 25 percent, a 5 percent reduction in thermal design flow and a removal of 65°F fuel temperature conservatism in the PAD fuel performance evaluation code. The reduction of thermal design flow was introduced to compensate for an additional hydraulic resistance caused by the plugged steam generator tubes. It is a conservative assumption. The removal of 65°F fuel temperature conservatism is a non-conservative assumption because in itself it would cause the peak cladding temperature to increase. However, other assumptions existing in the PAD code compensate for it and as a result the fuel performance evaluation by the code is conservative. This change has been approved by us in Reference 3.

The LOCA analysis was performed using the February 1978 version of the Westinghouse Evaluation Model (Reference 4) which was reviewed and approved by us (Reference 5). It was performed for a double ended cold leg guillotine break (DECLG) with a discharge coefficient of C_D = 0.4. The licensee has shown in the previous submittal (Reference 6) that this break size corresponds to the highest value of peak cladding temperature and Zr-water reaction. The licensee has also demonstrated that the break size remains unaffected by the number of the steam generator tubes plugged (Reference 7).

The previous analysis for Units 3 and 4 (Reference 8) was performed with the same evaluation model, assuming the same steam generator tube plugging level, but using $F_Q = 1.97$. However, a recently discovered input error required a new reanalysis of LOCA.

The currently submitted LOCA analysis includes the input corrections but it does not include the changes caused by the modified fuel performance models. The input parameters assumed in the analysis are listed below:

Core Power: 102 percent of 2200 MWt (rated power)
Peak Linear Power: 102 percent of 10.97 KW/ft
Peaking Factor: 1.93
Accumulator Water Volume: 875 cu ft/each

The results of the analysis indicate a peak cladding temperature of 2136°F, a maximum local Zr-water reaction of 6.945 percent and a total Zr-water reaction of less than 0.3 percent. All these values are below the limits specified in 10 CFR 50.46.

The licensee did not include small break analysis since steam generator tubes plugged did not affect significantly the results of the original analysis.

The licensee has provided additional calculations (Reference 1) to assess the potential impact of the recent concerns related to the fuel performance model changes included in draft report NUREG-0630 (Reference 2). Adoption of these changes would produce an increase of the peak cladding temperature by 405°F, due to the fuel burst model change and by 450°F, due to the fuel strain model change. To compensate for these changes and keep the peak cladding temperature below the 2200°F limit, the peaking factor F_Q should be reduced by 0.055. There are, however, two compensating effects which could provide credits offsetting the above mentioned penalties in LOCA analysis. These effects are due to the changes involving the slip and break flow models which have been approved by us for UHI plants after an extensive review. It is estimated that the total benefit of use of these models would be an increase of 0.38 units in F_Q. However, at the present moment, no adequate basis exists for considering horizontal slip. Also an uncertainty exists in translating the phenomena at blowdown to an effect during reflood. It is our current best technical judgment that application of these model changes would result in an increase of F_Q by 0.15 (Reference 9). This value more than offsets the penalties in F_Q and the results of the LOCA analysis submitted by the licensee (Reference 1) could be considered conservative.

The licensee has performed the "18 case FAC analyses" for Unit 3, Cycle 7 and Unit 4, Cycle 6 (Reference 10) because the limiting peaking factor in the LOCA analysis was below the value for which the excore detectors could give reliable measurements. The results of these analyses have indicated that for both units the predicted maximum peaking factor exceeds the limiting value of F_Q. The licensee is therefore required either to limit power to the rated power multiplied by the ratio of 1.93 divided by the predicted peaking factor or to implement the augmented surveillance discussed in Reference 11 and ascertain that the peaking factor would not exceed the limiting value of 1.93. This requirement could be lifted anytime during plant operation if the licensee demonstrates by the "18 case FAC analysis" that the maximum predicted F_Q is within the LOCA determined limit.

Summary

Based on the review of the submitted documents, we conclude that the results of the LOCA analysis performed with F_Q = 1.93 are conservative relative to the 10 CFR 50.46 criteria. We consider the resultant changes to the Technical Specifications acceptable for operating Units 3 and 4 with up to a maximum of 25 percent of steam generator tubes plugged.

Environmental Consideration

We have determined that the amendments do 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 amendments involve 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 these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date:

References

1. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), Serial No. L-80-171, dated June 5, 1980.

2. NUREG-0630, Cladding Swelling and Rupture Models for LOCA analysis, November 1979.

3. NRC Memo from P. S. Check to A. Schwencer, Safety Evaluation by NRR of LOCA Reanalysis for Zion Station, Units 1 and 2, dated March 14, 1980.

4. WCAP-9220, Westinghouse ECCS Evaluation Model, February 1978 Version, February 1978.

5. Letter from J. F. Stolz (NRC) to T. M. Anderson (Westinghouse), dated August 29, 1978.

6. Letter from R. E. Uhrig (FPL) to V. Stello (NRC), dated December 9, 1976.

7. Letter from R. E. Uhrig (FPL) to G. Lear (NRC), Serial No. L-77-217, dated July 11, 1977.

8. Letter from R. E.Uhrig (FPL) to D. G. Eisenhut (NRC), Serial No. L-80-129, dated April 29, 1980.

9. G. N. Lauben (NRC) to R. P. Denise (NRC) Memorandum, "Review Status of Considered Revisions to Vendor ECCS Evaluation Models," dated December 21, 1979.

10. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), Serial No. L-80-68 dated March 3, 1980.

11. Letter from R. E. Uhrig (FPL) to V. Stello (NRC, Serial No. L-78-127, dated April 10, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-250 AND 50-251

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 58 to Facility Operating License No. DPR-31, and Amendment No. 51 to Facility Operating License No. DPR-41 issued to Florida Power and Light Company (the licensee), which revised Technical Specifications for operation of Turkey Point Plant, Unit Nos. 3 and 4 (the facilities) located in Dade County, Florida. The amendments are effective as of the date of issuance.

The amendments incorporate the results of a revised ECCS analysis for a steam generator plugging level of 25% for both Unit 3 and Unit 4 and permit continued operation of Unit 4 for six equivalent months of operation from June 11, 1980, at which time the steam generators for Unit 4 shall be inspected.

The application for the amendments complies 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments 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 these amendments.

For further details with respect to this action, see (1) the applications for amendments dated March 14 (L-80-83), and June 5, 1980 (L-80-170 and L-80-171), (2) Amendment Nos. 58 and 51 to License Nos. DPR-31 and DPR-41, 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 Environmental and Urban Affairs Library, Florida International University, Miami, Florida 33199. A 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 Licensing.

Dated at Bethesda, Maryland, this 12th day of June, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing