Distribution
Docket
ORB #3
Local PDR
NRC PDR
VStello
KGoller
GLear
CParrish
DElliott
Attorney, OELD
OI&E (5)
BJones (8)
BSchark (10)
JMcGough
DEisenhut
ACRS (16)
OPA (Clare Miles)
DRoss
TBAbernthy
JRBuchanan

APR 29 1977

Dockets Nos. 50-250
and 50-251

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. DPR-31 and Amendment No. 24 to Facility Operating License No. DPR-41 for the Turkey Point Nuclear Generating Units Nos. 3 and 4. The amendments consists of changes to the Technical Specifications in response to your application dated December 9, 1976 and supplements dated December 9, December 30, 1976 and January 3, 1977.

The amendments concern changes to the Technical Specifications requested as a result of the emergency core cooling system (ECCS) cooling performance submitted in response to our Ordess for Modification of License dated August 27, and December 3, 1976. We are approving by this licensing action the reevaluation of the ECCS cooling performance as it applies to Turkey Point Unit No. 4. Since we previously approved, by letter dated January 14, 1977, the ECCS reevaluation as it applied to Turkey Point Unit No. 3, this licensing action completes our review of the ECCS cooling performance reevaluation for both Turkey Point Units Nos. 3 and 4. In addition, since you have fulfilled the requirements of our Orders of Modification of License dated August 27, and December 3, 1976, this licensing action removes the restrictions on total nuclear peaking factor (F_Q) specified in these orders.

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures and ccs:
See next page

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	CParrish	DElliott:mjf	S. Goldberg	GLear		
DATE	4/28/77	4/28/77	4/29/77	4/29/77		

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Enclosures:
1. Amendment No. 25 to License DPR-31
2. Amendment No. 24 to License DPR-41
3. Safety Evaluation
4. FEDERAL REGISTER NOTICE

cc:

Mr. Jack R. Newman, Esquire
Lowenstein, Newman, Reis & Axelrad
1025 Connecticut Avenue, N. W.
Suite 1214
Washington, D. C. 20036

Mr. Ed Maroney
Bureau of Intergovernmental Relations
725 South Bronough Street
Tallahassee, Flordia 32304

Honorable Dewey Knight
County Manager of Metropolitan
Dade County
Miami, Florida 33130

Florida Power & Light Company
ATTN: Mr. Henry Yaeger
Plant Manager
Turkey Point Plant
P. O. Box 013100
Miami, Florida 33101

Chief, Energy Systems Analysis Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region VI Office
ATTN: EIS COORDINATOR
345 Courtland Street, N. E.
Atlanta, Georgia 30308

Environmental & Urban Affairs Library
Florida International University
Miami, Florida 33199

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
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 December 9, 1976 (supplements dated December 9, December 30, 1976 and January 3, 1977), 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 25, 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

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: April 29, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 25

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-31

DOCKET NO. 50-250

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.

Remove	Insert
3.2-3	3.2-3
3.2-3a	3.4-1
3.4-1	B3.2-4
3.4-1a	B3.2-6
B3.2-4	
B3.2-4a	
B3.2-6	
B3.2-6a	

reactivity insertion upon ejection greater than 0.3% Δ 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. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_q(Z) \leq (2.22/P) \times K(Z)$ for $P > .5$

$F_q(Z) \leq (4.44) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]$

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 and Z is the core height location of F_q.

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 conform that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

3.4 ENGINEERED SAFETY FEATURES

Applicability: Applies to the operating status of the Engineered Safety Features.

Objective: To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, and (3) to remove airborne iodine from the containment atmosphere in the event of a Maximum Hypothetical Accident.

Specification: 1. SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS

a. The reactor shall not be made critical, except for low power physics tests, unless the following conditions are met:

1. The refueling water tank shall contain not less than 320,000 gal. of water with a boron concentration of at least 1950 ppm.

2. The boron injection tank shall contain not less than 900 gal. of a 20,000 to 22,500 ppm boron solution. The solution in the tank, and in isolated portions of the inlet and outlet piping, shall be maintained at a temperature of at least 145F. TWO channels of heat tracing shall be operable for the flow path.

3. Each accumulator shall be pressurized to at least 600 psig and contain 875-891 ft^3 of water with a boron concentration of at least 1950 ppm, and shall not be isolated.

4. FOUR safety injection pumps shall be operable.

An upper bound envelope of 2.22 times the normalized peaking factor axial dependence of Figure 3.2-3 has been determined to be consistent with the technical specifications on power distribution control as given in Section 3.2.

When an F_q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate experimental uncertainty allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_q, (b) the operator has a direct influence on F_q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of start-up physics tests, at least once each full rated power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear

Flux Difference ($\Delta\phi$) and a reference value which corresponds to the full design power equilibrium value of Axial Offset (Axial Offset = $\Delta\phi$/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the F_q upper bound envelope of 2.22 times Figure 3.2-3 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal rated power operating position appropriate for the time in life. Control rods are usually withdrawn farther as burnup proceeds). This value, divided by the fraction of design power at which the core was operating is the design power value of the target flux difference. Values for all other core power levels are obtained by multiplying the design power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of $\pm 5\%$ ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every rated power month. For this reason, methods are permitted by Item 6c of Section 3.2 for updating the target flux differences. Figure B3.2-1 shows a typical construction of the target flux difference band at BOL and Figure B3.2-2 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during the required, periodic excore calibra-

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. DPR-41

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated December 9, 1976 (supplements dated December 9, December 30, 1976 and January 3, 1977), 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-41 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 24, 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

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 29, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 24

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-251

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.

Remove	Insert
3.2-3	3.2-3
3.2-3a	3.4-1
3.4-1	B3.2-4
3.4-1a	B3.2-6
B3.2-4	
B3.2-4a	
B3.2-6	
B3.2-6a	

reactivity insertion upon ejection greater than 0.3% Δ 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. <u>CONTROL ROD POSITION INDICATION</u>

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. <u>POWER DISTRIBUTION LIMITS</u>

a. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_q(Z) \leq (2.22/P) \times K(Z)$ for $P > .5$

$F_q(Z) \leq (4.44) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ [1 + 0.2\ (1-P)]$

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 and Z is the core height location of F_q.

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 conform that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

3.4 ENGINEERED SAFETY FEATURES

Applicability: Applies to the operating status of the Engineered Safety Features.

Objective: To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, and (3) to remove airborne iodine from the containment atmosphere in the event of a Maximum Hypothetical Accident.

Specification: 1. SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS

a. The reactor shall not be made critical, except for low power physics tests, unless the following conditions are met:

1. The refueling water tank shall contain not less than 320,000 gal. of water with a boron concentration of at least 1950 ppm.

2. The boron injection tank shall contain not less than 900 gal. of a 20,000 to 22,500 ppm boron solution. The solution in the tank, and in isolated portions of the inlet and outlet piping, shall be maintained at a temperature of at least 145F. TWO channels of heat tracing shall be operable for the flow path.

3. Each accumulator shall be pressurized to at least 600 psig and contain 875-891 ft^3 of water with a boron concentration of at least 1950 ppm, and shall not be isolated.

4. FOUR safety injection pumps shall be operable.

An upper bound envelope of 2.22 times the normalized peaking factor axial dependence of Figure 3.2-3 has been determined to be consistent with the technical specifications on power distribution control as given in Section 3.2.

When an F_q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate experimental uncertainty allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_q, (b) the operator has a direct influence on F_q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of start-up physics tests, at least once each full rated power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear

Flux Difference ($\Delta\phi$) and a reference value which corresponds to the full design power equilibrium value of Axial Offset (Axial Offset = $\Delta\phi$/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the F_q upper bound envelope of 2.22 times Figure 3.2-3 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal rated power operating position appropriate for the time in life. Control rods are usually withdrawn farther as burnup proceeds). This value, divided by the fraction of design power at which the core was operating is the design power value of the target flux difference. Values for all other core power levels are obtained by multiplying the design power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of $\pm 5\%$ ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every rated power month. For this reason, methods are permitted by Item 6c of Section 3.2 for updating the target flux differences. Figure B3.2-1 shows a typical construction of the target flux difference band at BOL and Figure B3.2-2 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during the required, periodic excore calibra-

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO LICENSE NO. DPR-31, AND

AMENDMENT NO. 24 TO LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNITS NOS. 3 AND 4

DOCKETS NOS. 50-250 AND 50-251

Introduction

By letters dated December 9, 1976, Florida Power and Light Company (FPL) submitted an emergency core cooling system (ECCS) cooling performance reevaluation for Turkey Point Units Nos. 3 and 4 and proposed changes to the Technical Specifications of Facility Operating Licenses DPR-31 and DPR-41 based on this ECCS reevaluation. The proposed Technical Specification changes would modify the Unit No. 4 operating limits on total nuclear peaking factor (F_Q) and accumulator water volume. Supplemental information relating to the ECCS reevaluation and proposed Technical Specification changes were supplied by FPL in their letters dated December 30, 1976 and January 3, 1977.

Turkey Point Unit No. 4 is presently operating under the conditions of our Orders dated August 27, and December 3, 1976. These Orders restricted F_Q and required the ECCS cooling performance reevaluation. Our Order of December 3, 1976, supplemented our Order of August 27, 1976 and restricted F_Q to a conservative value of 2.08 in order to compensate for: (1) a reported reactor vessel upper head water temperature in excess of that assumed in the previously approved ECCS analysis and (2) up to 7% plugged steam generator tubes. The higher upper head water temperature and the plugged steam generator tubes have the effect of increasing the calculated peak clad temperature in the event of a loss-of-coolant-accident (LOCA). The ECCS reevaluation which FPL submitted on December 9, 1976 included: (1) the effect of higher primary coolant temperature in the upper head region of the reactor pressure vessel and (2) the effect of up to 10% plugged steam generator tubes.

On December 21, 1976, FPL proposed changes to the Unit No. 3 Technical Specification limits on F_Q and accumulator water volume. These proposed changes were based on the ECCS reevaluation of December 9, 1976 and contained proposed modifications of Unit No. 3 operating limits identical to those proposed for Unit No. 4 on December 9, 1976. The proposed operating limits were approved for Unit No. 3 on January 14, 1977 (Amendment No. 22 to License No. DPR-31 and Amendment No. 21 to License No. DPR-41, dated January 14, 1977). However, similar operating limits were not approved for Unit No. 4, at that time, as modifications to the Unit No. 4 accumulator liquid measuring taps were necessary before the new accumulator water volume limits could be adopted for Unit No. 4. Since the Unit No. 4 accumulator liquid volume measuring taps have now been modified the operating limits for Unit No. 4 can be changed so that they are identical with those changes previously incorporated into the Technical Specifications for Unit No. 3.

Because Units Nos. 3 and 4 share joint Technical Specifications, the Technical Specifications for Unit No. 3 are being changed to reflect the proposed changes to the Unit No. 4 Technical Specifications. However, the operating limits for Unit No. 3 are unchanged by this licensing action.

Discussion

In response to our Orders for Modification of License dated August 27 and December 3, 1976, FPL submitted on December 9, 1976, an ECCS reevaluation applicable to the Turkey Point Nuclear Generating Unit No. 4. This ECCS reevaluation supercedes the previous ECCS evaluation submitted on March 10, 1975 and includes: (1) the effect of a primary coolant temperature in the upper head region of the reactor pressure vessel equal to the primary coolant hot leg temperature and (2) the effect of up to 10% plugged steam generator tubes. Based on the ECCS reevaluation and to maintain the maximum calculated peak clad temperature below 2200°F following a LOCA, FPL requested the following changes in the Technical Specifications: (1) a decrease in the previously specified limit on total nuclear peaking factor (F_Q) and (2) an increase in the previously specified minimum accumulator water volume. Operation of Unit No. 4 to the proposed operating limits would decrease the peak clad temperature in the event of a LOCA. FPL proposed the new Unit No. 4 operating limits based on the ECCS reevaluation to justify removal of the conservative limit on F_Q specified in our Orders of August 27 and December 3, 1976.

Evaluation

The ECCS cooling performance following a loss-of-coolant-accident (LOCA) was reevaluated by FPL using the following assumptions.

(1) A limiting value for the total nuclear peaking factor (F_Q) equal to 2.22.

(2) A primary coolant temperature in the upper head region of the reactor pressure vessel equal to the primary coolant hot leg temperature.

(3) A total of 10% plugged steam generator tubes.

(4) A minimum water volume in the accumulator of 875 cubic feet.

Assumptions (1) and (4) resulted in Technical Specification changes proposed by FPL and included in this licensing action. These changes: (1) reduce F_Q and increase the accumulator minimum water volume from the values presently specified in the Technical Specifications and (2) are conservative because plant operation within these limits will result in a decrease in the peak clad temperature following a LOCA. Assumption (2) conforms to our Orders of August 27, and December 3, 1976 which required an ECCS reevaluation using the primary coolant temperature in the upper head region of the reactor pressure vessel equal to the primary coolant hot leg temperature. Assumption (3) is conservative because: (1) the fraction of steam generator tubes presently plugged in Unit No. 4 is 7% and (2) the prior ECCS evaluation did not include the effect of plugged steam generator tubes.

FPL identified the worst case LOCA as a double-ended cold leg guillotine break with a discharge coefficient of 0.4. The ECCS cooling performance reevaluation predicted that the worst case LOCA would result in: (1) a peak clad temperature of 2190°F, (2) a maximum local metal-water reaction of 11.9% and (3) a total core wide metal-water reaction of less than 0.3%. Our review of the ECCS cooling performance supports the conclusion that: (1) the peak clad temperature following a LOCA will be less than 2200°F, (2) the maximum local metal-water reaction will be less than 17% and (3) the total core wide metal-water reaction will be less than 1%. Therefore, the calculated ECCS cooling performance for Turkey Point Unit No. 4 conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR 50.46(b) and is acceptable. We further conclude that the ECCS cooling performance reevaluation was calculated in accordance with an approved Westinghouse evaluation model and satisfies our Orders of August 27 and December 3, 1976. Therefore, the restriction on F_Q specified in these Orders can be removed. In addition, our evaluation supports the conclusion that the Technical Specification changes correctly incorporate operating limits based on the ECCS reevaluation into the Technical Specifications. Therefore, we conclude that the proposed Technical Specification changes are acceptable.

Environmental Considerations

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 are 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 changes 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 changes 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.

Dated: April 29, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-250 AND 50-251

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 25 and 24 to Facility Operating Licenses Nos. DPR-31 and DPR-41, respectively, issued to Florida Power and Light Company which revised Technical Specifications for operation of the Turkey Point Nuclear Generating Units Nos. 3 and 4, located in Dade County, Florida. The amendments are effective as of the date of issuance.

These amendments concern changes required as a result of a reevaluation of the emergency core cooling system for Turkey Point Unit No. 4. The emergency core cooling system reevaluation fulfills, for Unit No. 4, the requirements of the Commission's Orders for Modification of License dated August 27, 1976 and December 3, 1976. The operating limits for Unit No. 3 set forth in its Technical Specifications remain unchanged although the Unit No. 3 Technical Specifications will be modified to reflect the revisions to the Unit No. 4 Technical Specifications.

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 application for amendments dated December 9, 1976, as supplemented by letters dated December 9, December 30, 1976 and January 3, 1977, (2) Amendments Nos. 25 and 24 to Licenses 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 Documents Room, 1717 H Street, N. W., Washington, D. C. and at the Environmental & 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 Operating Reactors.

Dated at Bethesda, Maryland, this 29 day of April 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors