Docket

JAN 1 4 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. 22 to Facility Operating License No. DPR-31 and Amendment No. 21 to Facility Operating License No. DPR-41 for Turkey Point Nuclear Generating Units Nos. 3 and 4. The amendments consist of an added condition to the license for Turkey Point Unit No. 3 and changes to the Technical Specifications in response to your request dated December 21, 1976 and supplements dated December 9, December 22, December 30, 1976 and January 3, 1977.

The amendments concern changes required as a result of the Unit No. 3 steam generator repair and the reevaluation of the emergency core cooling system (ECCS) cooling performance. We have approved the reevaluation of ECCS cooling performance for Turkey Point Unit No. 3 which you submitted in response to our Order for Modification of License dated August 27, 1976. Our review of the ECCS reevaluation for Turkey Point Unit No. 4 has not been completed. Therefore, you are reminded that this licensing action does not modify the limit on the Unit No. 4 total nuclear peaking factor contained in our Order for Modification of License dated December 3, 1976.

We concur that the repair program for the steam generators of Turkey Point Unit No. 3 is adequate subject to the conditions of the amendment to the license of Turkey Point Unit No. 3. You are requested to submit the details of your steam generator inspection program no later than 30 days prior to the date you expect the next inspection to commence. Following the steam generator inspection, you shall: (1) provide information to justify continued steam generator operation and (2) obtain Nuclear Regulatory Commission approval before resuming power operation.

9 const 1

OFFICE➤						
SURNAME➤						
DATE➤						

JAN 1 4 1977

Copies of our related Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

Original signed by

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Enclosures:
1. Amendment No. 22 to License DPR-31
2. Amendment No. 21 to License DPR-41
3. Safety Evaluation
4. Federal Register Notice

cc: See next page

DISTRIBUTION:
Dockets
NRC PDR
Local PDR
ORB#3 Rdg
VStello
KRGoller/TJCarter
CParrish
DElliott
OELD
OI&E (5)
BJones (8)
BScharf (16)
JMcGough
JSaltzman
DEisenhut
ACRS (16)
CMiles
DRoss
TBAbernathy
JRBuchanan
File
Xtra Copies

OFFICE➤	ORB#3	ORB#3	ORB#4	OELD	ORB#3	AD/DOR
SURNAME➤	CParrish	DElliott:acr	MFairtile		GLear	KRGoller
DATE➤	1/ /77	1/ /77	1/ /77	1/ /77	1/ /77	1/ /77

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. 22
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 21, 1976, as supplemented by letters dated December 9, December 22, 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 adding a new Paragraph 3.E as follows:

   "E. Steam Generator Inspections

   In order to perform an inspection of the steam generators, Unit No. 3 shall be brought to the cold shutdown condition within six equivalent months of operation from January 14, 1977. Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection. Unit No. 3 operation up to six equivalent months of operation is authorized provided that further information and evaluations required in connection with continued operation of Unit No. 4, as a result of Amendment No. 20 to Facility Operating License DPR-41, justify continued operation of Unit No. 3 for the full period authorized by this paragraph 3.E.

For the purpose of this requirement, equivalent operation is defined as operation with a primary coolant temperature greater than 350°F."

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: January 14, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 22

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DRP-31

DOCKET NO. 50-250

Replace pages 3.2-3, 3.4-1, B3.2-4 and B3.2-6 with the attached revised pages. Add pages 3.2-3a, 3.4-1a, B3.2-4a and 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 $P > .5$

$F_q(Z) \leq (4.48) \times K(Z)$ 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 2.24 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-

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:

$F_q(Z) \leq (2.32/P) \times K(Z)$ for $P > .5$

$F_q(Z) \leq (4.64) \times K(Z)$ for $P \leq .5$

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

where P is the fraction of design 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 825-841 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 2.32 times the normalized peaking factor axial dependence of Figure 3.2-3 has been determined (from extensive analyses at design power considering all operating maneuvers) to be consistent with the technical specifications on power distribution control as given in Section 3.2. The results of the loss of coolant accident analyses based on this upper bound envelope indicate a peak clad temperature of 2150°F at design power, corresponding to a 50°F margin to the 2200°F FAC limit.

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 2.32 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. 21
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 licensees) dated December 21, 1976, as supplemented by letters dated 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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

Dated: January 14, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 21

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-251

Replace pages 3.2-3, 3.4-1, B3.2-4 and B3.2-6 with the attached revised pages. Add pages 3.2-3a, 3.4-1a, B3.2-4a and 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 $P > .5$

$F_q(Z) \leq (4.48) \times K(Z)$ 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 2.24 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-

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:

$F_q(Z) \leq (2.32/P) \times K(Z)$ for $P > .5$

$F_q(Z) \leq (4.64) \times K(Z)$ for $P \leq .5$

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

where P is the fraction of design 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 825-841 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 2.32 times the normalized peaking factor axial dependence of Figure 3.2-3 has been determined (from extensive analyses at design power considering all operating maneuvers) to be consistent with the technical specifications on power distribution control as given in Section 3.2. The results of the loss of coolant accident analyses based on this upper bound envelope indicate a peak clad temperature of 2150°F at design power, corresponding to a 50°F margin to the 2200°F FAC limit.

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 2.32 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. 22 TO LICENSE NO. DPR-31, AND

AMENDMENT NO. 21 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

Following an 80 gpm leak in a steam generator tube at the Virginia Electric and Power Company's (VEPCO) Surry Power Station Unit No. 2 on September 15, 1976, Florida Power and Light Company (FPL) participated in a joint inspection program designed to investigate the cracking of small bend radius steam generator tubes. By letter dated October 26, 1976, FPL provided the NRC with their steam generator tube investigation program for Turkey Point Units Nos. 3 and 4. FPL removed Turkey Point Unit No. 3 from service on November 14, 1976, for inspection of the steam generators and committed not to resume operation until the NRC concurred in the corrective action to be taken. By letter dated December 21, 1976, FPL informed the NRC of: (1) the results of the Unit No. 3 steam generator inspection and (2) their proposed corrective action, and requested NRC concurrence to return Unit No. 3 to power operation. Supplemental information relating to the steam generator inspection was supplied by FPL in their letters dated December 22, 1976, December 30, 1976, and January 3, 1977.

The corrective action proposed by FPL for Unit No. 3 will result in the number of plugged steam generator tubes increasing from less than 3% to approximately 7%. The increased number of plugged steam generator tubes increases the calculated predicted peak clad temperature in the event of a loss-of-coolant accident (LOCA). Prior to the shutdown for steam generator tube inspection, Unit No. 3 was operating under the conditions of an NRC Order for Modification of License dated August 27, 1976. This Order restricted the total nuclear peaking factor (F_Q) to 2.11 and required a submittal of an emergency core cooling system (ECCS) cooling performance reevaluation, as soon as possible. The effect of increasing the number of plugged steam generator tubes invalidates one of the assumptions upon which our Order of August 27, 1976 is based.

Therefore, in response to our Order of August 27, 1976, and in support of the Unit No. 3 steam generator corrective action, FPL submitted on December 9, 1976 an ECCS reevaluation which 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 Technical Specifications of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Nuclear Generating Units Nos. 3 and 4. The proposed changes are based on the ECCS reevaluation and contains a modification of the Unit No. 3 operating limits on total nuclear peaking factor (F_Q) and accumulator water volume. Supplemental information relating to the ECCS reevaluation and the proposed Technical Specification changes was supplied by FPL in their letters of December 30, 1976 and January 3, 1977.

Because Units 3 and 4 share joint Technical Specifications, FPL proposed modifying the Technical Specifications for Unit 4 to reflect the proposed revision to the Unit 3 Technical Specifications. However, the operating limits for Unit 4 are unchanged by this licensing action.

STEAM GENERATOR OPERATION

Discussion

By letters dated December 21, 22, 30, 1976 and January 3, 1977, FPL submitted results concerning the steam generator inspections at Turkey Point Unit No. 3. These letters relate to the steam generator tube "denting" phenomenon in Unit No. 3, the closure of the tube support plate flow slots, and additional information in response to the staff's concern over the cause and the consequences of continuing upper support plant deformation at the flow slot locations. The NRC requested this inspection because Turkey Point Units Nos. 3 and 4 and some other pressurized water reactors (PWRs) have experienced extensive denting of steam generator tubes. Tube denting has been identified as the initial condition that led to a large leak at the U-bend apex in a row 1 tube of the Surry Unit No. 2 steam generator "A". The denting phenomenon caused significant deformation of the top tube support plate at flow slot locations, in turn forcing an inward displacement of the steam generator tube U-bend legs. This inward movement of the U-bend legs caused an increase in the service strain at the U-bend apex sufficient to initiate and enhance the susceptibility to "intergranular cracking" of the Inconel 600 Alloy tubing exposed to the PWR primary coolant.

FPL has performed eddy current examinations of rows 1 through 5 around U-bends in all three steam generators of Turkey Point Unit No. 3, and measured the flow slot openings in the bottom and top tube support plates of all three steam generators. Tube support plate deformation of the flow slot locations in steam generator 3B is more advanced than steam generators 3A and 3C. The eddy current examinations found no defects at the U-bends in rows 1 through 5 in each steam generator.

Flow slot measurements taken at the bottom tube support plate in steam generator 3B revealed evidence of closure in all six flow shots. FPL has estimated the rate of flow slot closure to be 0.126 inch per calendar month at the bottom tube support plate, where the maximum slot displacement was 1-3/4 inches. These rates were determined from measurements taken in November and December 1975. The maximum flow slot displacement is 3/4 inches in the top support plate. Therefore a more realistic flow slot closure rate in the top support plate would be 0.036 inches per calendar month. This closure rate is based on the ratio of the top and bottom support plate flow slot displacement times the bottom support plate closure rate.

Westinghouse has examined 71 tubes removed from rows 1, 2 and 3 from the Surry Units Nos. 1 and 2 and Turkey Point Unit No. 4 steam generators. Evidence of intergranular cracking at the U-bend apex was found only in the row 1 tubes at flow slot locations.

FPL also provided calculations to show that, even if the flow slots were to close completely, the total effective strain at the U-bend apex for all tubes in rows 2 through 4 or beyond would not reach the level calculated for row 1 tubes which have the smallest bend radius. The total effective strain consists of manufacturing strains, service strains, and the additional service strains due to the deformation of the top support plate at the flow slots. On the basis of a comparison of total effective strains, the results of laboratory examinations, and the relatively slow closure rate of the flow slots at the top support plantes, FPL concluded that tubes in rows 2 through 4 and beyond with larger U-bend radii will not be strained to the level for the initiation of intergranular stress corrosion cracking. Upon removing all row 1 tubes in all three steam generators from service by plugging, FPL concluded that continuing operation for approximately tne (10) months to complete the next fuel cycle is justified. Therefore the licensee proposed to resume power operation on January 10, 1977.

Evaluation

The licensee has submitted both analytical and experimental data in support of the corrective action for Turkey Point Unit No. 3 and the return to power operation. We have reviewed these data and have performed independent evaluations to determine the adequacy of the corrective action and the continued operation of Turkey Point Unit No. 3.

Regarding the tube plugging criteria applied to Turkey Point Unit No. 3 steam generators, the tube "denting" phenomenon, and the potential for "intergranular cracking" at the U-bend apex of the tubes in rows 2 through 5, we have considered the following issues in our safety evaluation of Turkey Point Unit No. 3: (1) the strain in the steam generator tubes at the U-bend apex is displacement controlled by the top tube support plate deformation at flow slot locations, (2) the deformation of the top support plate at the flow slots will not cause significant additional strain at the U-bend apex of the tubes in rows 2 through 5, during the next fuel cycle operation, (3) all of row 1 is plugged and the likelihood for the initiation of intergranular cracking of the unplugged tubes in rows 2 and beyond is minimal over the next operational period, (4) although tube "denting" is associated with tube support plate corrosion, support plate cracking, and the deformation of the tube support plate at flow slots, it does not reflect an immediate concern regarding tube integrity of the Turkey Point Unit No. 3 steam generators, because there have been relatively few leaks at the dent locations and no rapid failures have occurred, (5) no cracking has been observed in any tubes from rows 2 and outward, (6) the total effective strain at the U-bend apex anticipated for the unplugged tubes in rows 2 through 5, as a result of continued deformation of the top support plate at flow slots, will be substantially less than that incurred in the row 1 tubes, and row 1 has been plugged.

In addition, on the basis of the analytical and experimental data and the examination of 71 tubes removed from Surry Units Nos. 1 and 2 and from Turkey Point Unit No. 4, the staff concurs that it is unlikely that cracking at the U-bend apex for tubes in rows 2 and outward would occur in the Turkey Point Unit No. 3 steam generators during normal operation or postulated accidents, in the next fuel cycle. However, in order to obtain more complete flow slot closure data, an inspection within six months should be conducted. In addition the staff will consider the additional information obtained in connection with Unit No. 4 in their continuing review of Unit No. 3 operation.

We therefore conclude that:

1. Tubes in rows 2 through 5 and outward in all the steam generators of Turkey Point Unit No. 3 will retain sufficient integrity to withstand normal operating and postulated accident conditions.

2. There is reasonable assurance of tube integrity to provide adequate protection to the public health and safety.

3. Turkey Point Unit No. 3 should be inspected within 6 months to assess the magnitude and consequences of tube support plate deformation.

4. Turkey Point Unit No. 3 operation beyond the 6 month interval shall be dependent on the staff's ongoing evaluation of forthcoming information from facilities with denting and deformation of the tube support plates.

EMERGENCY CORE COOLING SYSTEM (ECCS)

Discussion

In response to our Order for Modification of License dated August 27, 1976, FPL submitted on December 9, 1976, an ECCS reevaluation applicable to the Turkey Point Nuclear Generating Unit No. 3. This ECCS reevaluation supercedes the previous ECCS evaluation submitted on March 10, 1975. Based on the ECCS reevaluation and to maintain the maximum calculated peak clad temperature below 2200°F, FPL requested on December 21, 1976 the following changes in the Technical Specifications; (1) a decrease in the limit on total nuclear peaking factor (F_Q) and (2) an increase in accumulator water volume.

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 calculated peak clad temperature following a LOCA. Assumption (2) conforms to our Order of August 27, 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. 3 is less than 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. 3 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 Order of August 27, 1976. Therefore, the restriction on total nuclear peaking factor (F_Q) to 2.11 specified in our Order of August 27, 1976 can be removed.

ENVIRONMENTAL FINDING

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: January 14, 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

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 22 and 21 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 (1) the steam generator repair and (2) a reevaluation of the emergency core cooling system, for Turkey Point Unit No. 3. The emergency core cooling system reevaluation fulfills, for Unit No. 3, the requirements of the Commission's Order for Modification of License dated August 27, 1976. The operating limits for Unit No. 4 set forth in its Technical Specifications remain unchanged although the Unit No. 4 Technical Specifications will be modified to reflect the revisions to the Unit No. 3 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 21, 1976, as supplemented by letters dated December 9, December 22, December 30, 1976 and January 3, 1977, (2) Amendments Nos. 22 and 21 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 Document 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 14th day of January 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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