JUN 5 1975

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. 9 to Facility Operating License No. DPR-31 and Amendment No. 8 to Facility Operating License No. DPR-41 for Turkey Point Nuclear Generating Units 3 and 4. These amendments include Change No. 21 to the joint Technical Specifications and are in response to your requests dated September 27, 1974, February 10, 1975, and March 11, 1975, and Supplements dated February 7 and 13, March 10, April 10 and 30 and May 13, 15, and 21, 1975.

These amendments: (1) incorporate operating limits in the Technical Specifications for the facilities based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR 50, and (2) modify certain Unit 4 operating limits to reflect the results of the cycle 2 core performance analysis.

The Commission's staff has evaluated the potential for environmental impact associated with operation of the Turkey Point Nuclear Generating Units 3 and 4 in the proposed manner. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no increase in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Part 51, Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

You will note that the Technical Specifications: (1) allow reactor operation only with three reactor coolant pumps in operation, and (2) require that the power supply be removed from the operators of certain specified valves in the Emergency Core Cooling System.

A copy of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 9
2. Amendment No. 8
3. Negative Declaration
4. Environmental Impact Appraisal
5. Safety Evaluation
6. Federal Register Notice

cc w/enclosures:
See next page

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
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 September 27, 1974, February 10 and March 11, 1975, and Supplements dated February 7 and 13, March 10, April 10 and 30, and May 13, 15, and 21, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility 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, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 21."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 21 to the
Technical Specifications

Date of Issuance: JUN 5 1975

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 8
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 September 27, 1974, February 10 and March 11, 1975, and Supplements dated February 7 and 13, March 10, April 10 and 30, and May 13, 15, and 21, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility 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, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 21."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 21 to the
Technical Specifications

Date of Issuance: JUN 5 1975

ATTACHMENT TO LICENSE AMENDMENTS NOS. 9 AND 8

CHANGE NO. 21 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSES NOS. DPR-31 AND DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert New Pages
i	i
1-4 - 1-6	1-4 - 1-6
2.3-2	2.3-2
	2.3-2a
2.3-3	2.3-3
	2.3-3a
3.2-1	3.2-1
3.2-3 - 3.2-6	3.2-3 - 3.2-8
Figures 3.2-1 & 3.2-1(a)	Figures 3.2-1 & 3.2-1(a)
Figure 3.2-3	Figure 3.2-3
3.4-1 - 3.4-2	3.4-1 - 3.4-2(a)
B3.2-1	B3.2-1
B3.2-3 - B3.2-6	B3.2-3 - B3.2-8
	Figures B3.2-1 & B3.2-2

TABLE OF CONTENTS

is subcritical, by an amount greater than or equal to the margin as specified in Figure 3.2-2 and T_{avg} is above 540F.

3) Refueling Shutdown

The reactor is in the refueling shutdown condition when the reactor is subcritical by at least 10% $\Delta k/k$ and T_{avg} is below 160F. A refueling shutdown refers to a scheduled shutdown to replace and shuffle fuel.

1.9 POWER OPERATION

The reactor is at power operation when it is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

1.10 REFUELING OPERATION

Refueling operation is any operation involving movement of fuel within the reactor vessel.

1.11 RATED POWER (R.P.)

Rated power is the licensed steady state reactor core thermal power output of 2200 MWt.

1.12 THERMAL POWER

Thermal power is the total core heat transferred from the fuel to the coolant.

1.13 DESIGN POWER

Design Power is the steady state reactor core thermal output of 2300 MWt. 21

1.14 ABNORMAL OCCURRENCE 21

An abnormal occurrence is defined as any of the following:

1. A safety system setting less conservative than the limiting setting established in the Technical Specifications.

2. Violation of a limiting condition for operation established in the Technical Specifications.

3. An uncontrolled or unplanned release of radioactive material from any plant system designed to act as a boundary for such material in an amount of significance with respect to limits prescribed in Technical Specification 3.9.

4. Failure of a component of an engineered safety feature or safety system that causes or threatens to cause the feature or system to be incapable of performing its intended function. Simultaneous failure of more than one component making up a redundant system shall be considered a failure under this definition. In addition, any failure of a component of an engineered safety feature or safety system shall be considered a failure under this definition unless it can be shown that the fault was not generic in nature.

5. Abnormal degradation of one of the several boundaries designed to contain the radioactive materials resulting from the fission process.

6. Significant (greater than 1% $\Delta k/k$) uncontrolled or unanticipated changes in reactivity.

7. Observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the plant.

8. Conditions arising from natural or offsite manmade events that affect or threaten to affect the safe operation of the plant. 21

1.15 POWER TILT

The power tilt is the ratio of the maximum to average of the upper out-of-core normalized detector currents or the lower out-of-core normalized detector currents whichever is greater. If one out-of-core detector is out of service, the remaining three detectors are to be used to compute the average.

1.16 INTERIM LIMITS

1.16.1 Fuel Residence Time Limit

The fuel residence time for Unit 3 shall be limited to 23,000 effective full power hours (EFPH) under low pressure operating conditions. The fuel residence time for Unit 4 shall be limited to 30,000 EFPH.

1.16.2 Reactor Coolant Pumps Operation

The reactor shall not be operated with less than three 21
reactor coolant pumps in operation.

1.17 LOW POWER PHYSICS TESTS

Low power physics tests are tests below a nominal 5% of rated power which measure fundamental characteristics of the reactor core and related instrumentation.

Reactor Coolant Temperature

Overtemperature ΔT $\leq \Delta T_o \left[K_1 - 0.0174(T-566.6) + 0.000976(P-1885) - f(\Delta q) \right]$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

P = Pressurizer pressure, psig

$f(\Delta q)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during startup tests such that:

For $(q_t - q_b)$ within +10 percent and -14 percent where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, $f(\Delta q) = 0$.

For each percent that the magnitude of $(q_t - q_b)$ exceeds +10 percent, the Delta-T trip set point shall be automatically reduced by 3.5 percent of its value at interim power.

For each percent that the magnitude of $(q_t - q_b)$ exceeds -14 percent, the Delta-T trip set point shall be automatically reduced by 2 percent of its value at interim power.

K_1 (Three Loop Operation) = 1.120;
(Two Loop Operation) = 0.88

Reactor Coolant Temperature

Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - 0.0107 \ (T - 574) + 0.000453 \ (P-2235) - f(\Delta q) \right]$

ΔT_o = Indicated ΔT at rated power

T = Average temperature, F

P = Pressurizer pressure, psig

$f(\Delta q)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during startup tests such that:

For $(q_t - q_b)$ within +10 percent and -14 percent 21
where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, $f(\Delta q) = 0$.

For each percent that the magnitude of $(q_t - q_b)$ exceeds +10 percent, the Delta-T trip set point shall be automatically reduced by 3.5 percent of its value at interim power.

For each percent that the magnitude of $(q_t - q_b)$ exceeds -14 percent, the Delta-T trip set point shall be automatically reduced by 2 percent of its value at interim power.

K_1 (Three Loop Operation) = 1.095;
(Two Loop Operation) = 0.88

Reactor Coolant Temperature

Overtemperature ΔT $\leq \Delta T_o \left[K_1 - 0.0174(T-566.6) + 0.000976(P-1885) - f(\Delta q) \right]$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

P = Pressurizer pressure, psig

$f(\Delta q)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during startup tests such that:

For $(q_t - q_b)$ within +10 percent and -14 percent where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, $f(\Delta q) = 0$.

For each percent that the magnitude of $(q_t - q_b)$ exceeds +10 percent, the Delta-T trip set point shall be automatically reduced by 3.5 percent of its value at interim power.

For each percent that the magnitude of $(q_t - q_b)$ exceeds -14 percent, the Delta-T trip set point shall be automatically reduced by 2 percent of its value at interim power.

K_1 (Three Loop Operation) = 1.120;
(Two Loop Operation) = 0.88

Overpower $\Delta T \leq \Delta T_o \left[1.09 - K_1 \frac{dT}{dt} - K_2 (T - T') - f(\Delta I) \right]$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

T' = Indicated average temperature at nominal conditions and rated power, F

K_1 = 0 for decreasing average temperature, 0.2 sec./F for increasing average temperature

K_2 = 0.00134 for T equal to or more than T'; 0 for T less than T'

$\frac{dT}{dt}$ = Rate of change of temperature, F/sec

$f(\Delta I)$ = As defined above

Pressurizer

Low Pressurizer pressure - equal to or greater than 1715 psig.

High Pressurizer pressure - equal to or less than 2385 psig.

High Pressurizer water level - equal to or less than 92% of full scale.

Reactor Coolant Flow

Low reactor coolant flow - equal to or greater than 90% of normal indicated flow

Low reactor coolant pump motor frequency - equal to or greater than 56.1 Hz

Under voltage on reactor coolant pump motor bus - equal to or greater than 60% of normal voltage

Steam Generators

Low-low steam generator water level - equal to or greater than 5% of narrow range instrument scale

~~Corrected~~ Revised page submitted on 8-8-75

Overpower $\Delta T \leq \Delta T_o \left[1.11 - K_1 \frac{dT}{dt} - K_2 (T - T') - f(\Delta q)\right]$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

T' = Indicated average temperature at nominal conditions and rated power, F

K_1 = 0 for decreasing average temperature, 0.2 sec./F for increasing average temperature

K_2 = 0.00068 for T equal to or more than T'; 0 for T less than T'

$\frac{dT}{dt}$ = Rate of change of temperature, F/sec.

$f(\Delta q)$= As defined above

Pressurizer

Low Pressurizer pressure - equal to or greater than 1835 psig.

High Pressurizer pressure - equal to or less than 2385 psig.

High Pressurizer water level - equal to or less than 92% of full scale.

Reactor Coolant Flow 21

Low reactor coolant flow - equal to or greater than 90% of normal indicated flow.

Low reactor coolant pump motor frequency - equal to or greater than 56.1 Hz.

Under voltage on reactor coolant pump motor bus - equal to or greater than 60% of normal voltage.

Steam Generators

Low-low steam generator water level - equal to or greater than 5% of narrow range instrument scale.

3.2 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability: Applies to the operation of the control rods and power distribution limits.

Objective: To ensure (1) core subcriticality after a reactor trip, (2) a limit on potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification: 1. CONTROL ROD INSERTION LIMITS

a. Whenever the reactor is critical, except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn.

b. Whenever the reactor is critical, except for physics tests and control rod exercises, the control group rods shall be no further inserted than the limits shown by the solid lines on Figure 3.2-1 for three loop operation and on Figure 3.2-1(a) for two loop operation. 2

c. After 70% of the second and subsequent cycles as defined by burnup, the limits shall be adjusted as a linear function of burnup toward the end-of-core life as shown by the dotted lines on Figure 3.2-1.

d. The end-of-core life limit shown on Figure 3.2-1 may be revised on the basis of physics calculations and physics data obtained during startup and subsequent operation.

e. Part length rods shall not be permitted in the core except for low power physics tests and for axial offset calibration tests performed below 75% of rated power.

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 21
   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,

(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, $F_{\Delta H}^{N}$, 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 $F_{\Delta H}^{N}$ 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. 2:

c. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least 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 difference 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 from its target band, the flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

f. At a power level no greater than 90% and above 50% of rated power,

   1. The indicated axial flux difference may deviate from its ±5% target band for a maximum of sixty effective minutes (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -11% and +11% at 90% of rated power and increasing by -1% and +1% for each 2% of rated power below 90% rated power.
      Effective time out of the target band is defined as the sum of the time out of the target band at power levels above 50% plus one half the time out of the target band at power levels of 50% and below. 21

   2. If item 1 above is violated, then the reactor power shall be reduced to no greater than 50% of rated power and the high neutron flux setpoint reduced to no greater than 55% of rated power.

   3. A power increase to a level greater than 90% of rated power is contingent upon the indicated axial flux difference being within its traget band.

g. At a power level no greater than 50% of rated power,

   1. The indicated axial flux difference may deviate from its target band.

   2. A power increase to a level greater than 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than sixty effective minutes (cumulative) out of the preceeding 24 hour period.

h. If the quadrant to average power tilt exceeds a value of 2% except for physics and rod exercise testing, then: 2

1) The hot channel factors shall be determined within 2 hours and the power level and trips adjusted to meet the requirements of Item 6a.

2) If the hot channel factors are not determined within two hours, the power shall be reduced from rated power 2% for each percent of quadrant tilt.

3) If the quadrant to average power tilt exceeds $\pm 10\%$, except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt. 21

i. If after a further period of 24 hours, the power tilt in k. above is not corrected to less than +2%, and

1) If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as an abnormal occurrence to the Nuclear Regulatory Commission.

2) If the hot channel factors are not determined, the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

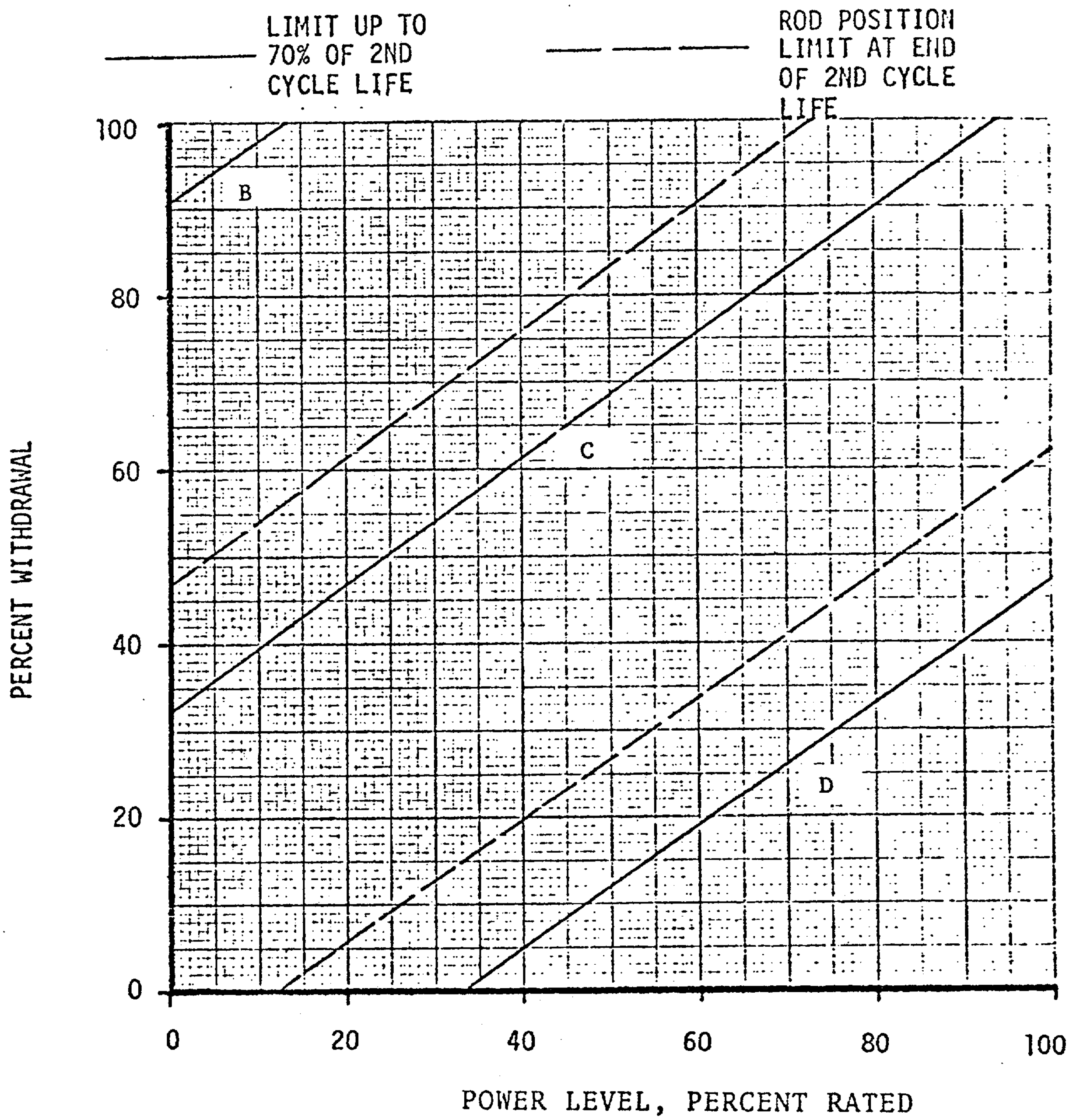
7. <u>IN-CORE INSTRUMENTATION</u>

a. A minimum of 16 thimbles, at least 2 per quadrant, and the necessary associated detectors shall be operable during the check and calibration of nuclear instrumentation ion chambers.

b. Power shall be limited to 90% of rated power for 3 loop or 50% of rated power for 2 loop operation if the requirements in Section 7.a are not met.

8. AXIAL OFFSET ALARMS

Alarms are provided to indicate non-conformance with the flux difference requirement 3.2.6e and the flux difference time requirement of 3.2.6f. If the alarms are temporarily out of service, conformance 2 with the applicable limit and the flux difference shall be logged at hourly intervals for the first twenty-four hours and half-hourly thereafter.



21

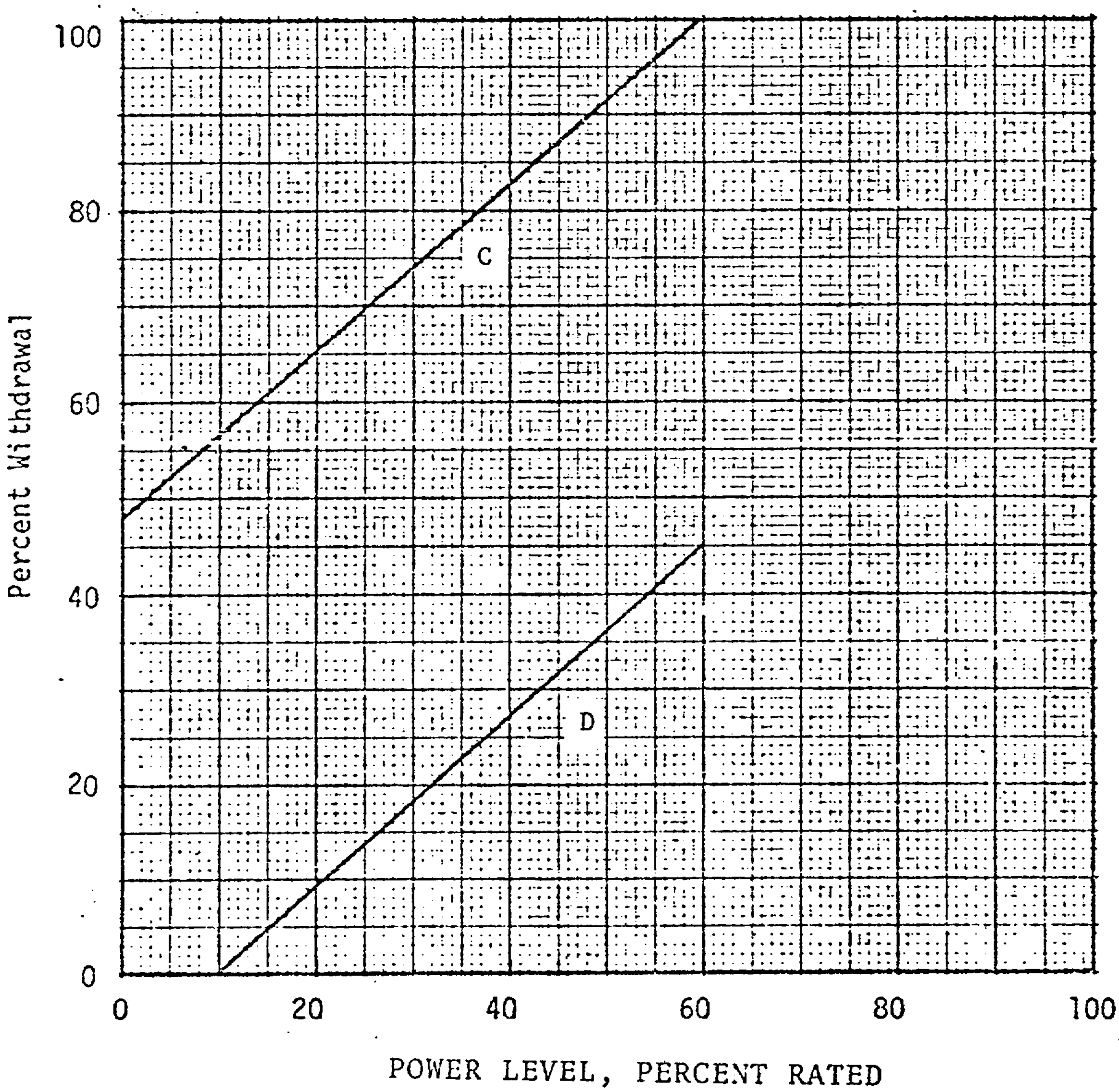
CONTROL GROUP INSERTION LIMITS
FOR THREE LOOP OPERATION

21

FIGURE 3.2-1

FIGURE 3.2-1(a)

CONTROL GROUP INSERTION LIMITS
FOR CYCLE 2
TWO LOOP OPERATION 21



21

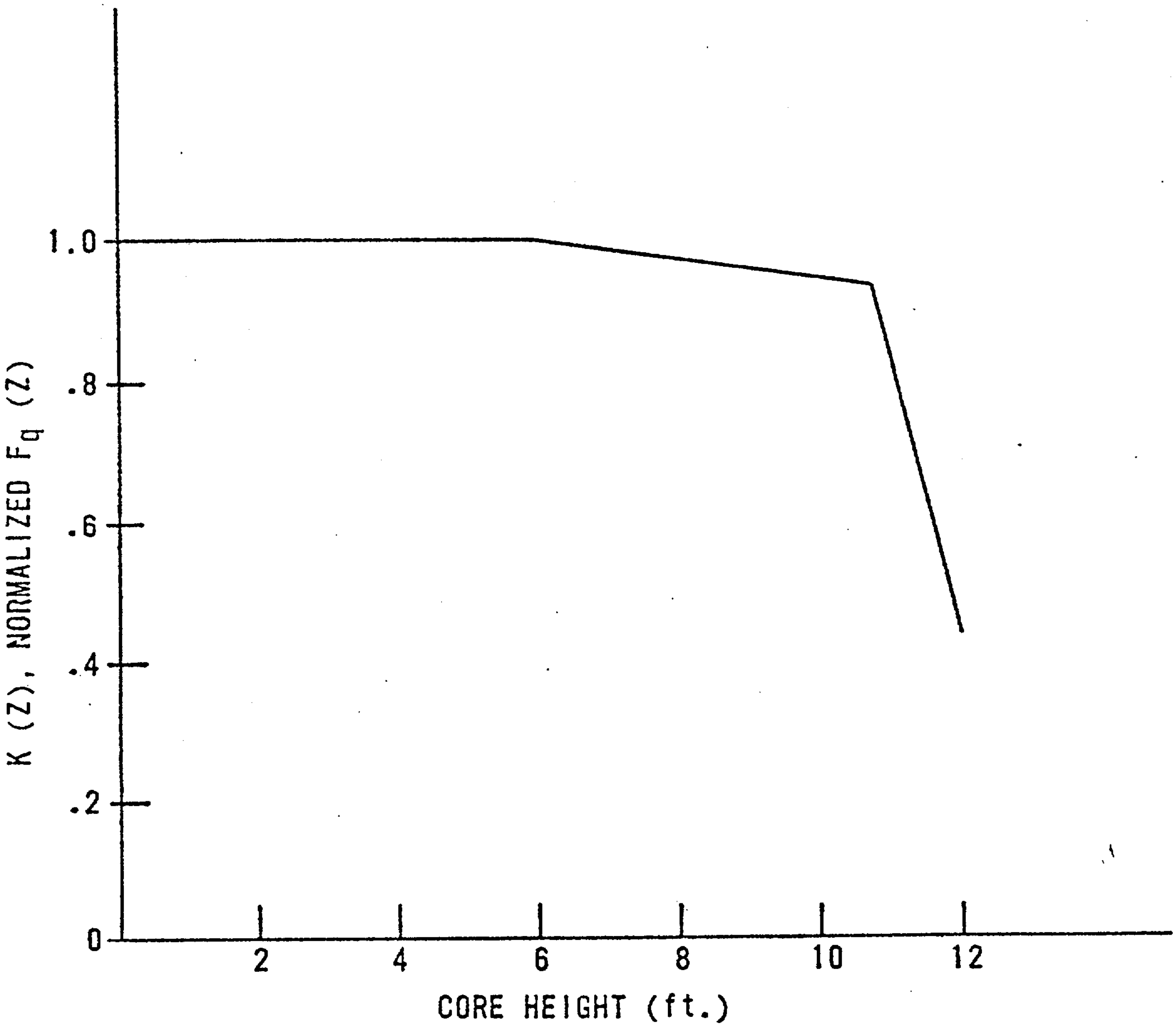


Figure 3.2-3 HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

21

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.

5. TWO residual heat removal pumps shall be operable
6. TWO residual heat exchangers shall be operable
7. All valves, interlocks and piping associated with the above components and required for post accident operation, shall be operable, except valves that are positioned and locked. Valves 864-A, B; 862-A, B; 865-A, B, C; 866-A, B shall have power removed from their motor operators by locking open the circuit breakers at the Motor Control Centers. The air supply to valve 758 shall be shut off to the valve operator. 21

b. During power operation, the requirements of 3.4.1a may be modified to allow one of the following components to be inoperable (including associated valves and piping) at any one time except for the cases stated in 3.4.1.b.2. If the system is not restored to meet the requirements of 3.4.1a within the time period specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 3.4.1a are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition.

1. ONE accumulator may be out of service for a period of up to 4 hours.

2. ONE of FOUR safety injection pumps may be out of service for 30 days. A second safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours. TWO of the FOUR safety injection pumps shall be tested to demonstrate operability before initiating maintenance of the inoperable pumps.

3. ONE channel of heat tracing on the flow path may be out of service for 24 hours.

4. ONE residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours. In addition the other residual heat removal pump shall be tested to demonstrate operability prior to initiating maintenance of the inoperable pump.

B3.2 BASES FOR LIMITING CONDITIONS FOR OPERATION, CONTROL AND POWER DISTRIBUTION LIMITS

Reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis.[1] In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and provide for acceptable nuclear peaking factors. The solid line shown on Figure 3.2-1 meets the shutdown requirement for the first 70% of second and subsequent cycles for Units 3 and 4, except for two loop operation. The end-of-core-life limit may be more restrictive, as shown by the conservative 21
estimate represented by the dotted line. The end-of-core-life limit may be determined on the basis of startup and operating data to provide a more realistic limit which will allow for more flexibility in operation and still assure compliance with the shutdown requirement. Figure 3.2-1(a) shows the shutdown requirements for second cycle two loop operation. The maximum shutdown margin requirement occurs at end-of-core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 3.2-2 shows the shutdown margin equivalent to 1.77% reactivity at end-of-core-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

The overlap between successive control banks is allowed because the control rod worth is lower near the top and bottom of the core than in the center.

Positioning of the part-length rods is governed by the requirement to maintain the axial power shape within specified limits or to accept an automatic cutback of the overpower ΔT and overtemperature ΔT set points (see Specification 2.3). Thus, there is no need for imposing a limit on the physical positioning of the part-length rods.

Design criteria have been chosen for normal and operating transient events which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to conditions imposed for normal and operating transient events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS Acceptance Criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution, the following hot channel factors are defined.

$F_q(Z)$, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux. 2

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

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 uncertain- 21
ties 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 design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which could, otherwise, affect these bases. For normal operation, it is not necessary to measure these quantities. Instead, it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows.

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment alarm of 12 steps precludes a rod misalignment greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks.

3. The full length control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core. 21

The permitted $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In specification 3.2, F_q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of

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 calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibration. This is acceptable due to the extremely low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band. However, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are 2
observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14% to -14% (+11% to -11% indicated) increasing by $\pm$ 1% for each 2% decrease in rated power. Therefore, while the deviation exists, the power level is limited to 90% of design power or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the $\pm$5% band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50% of design power is required to protect against potentially more severe consequences of some accidents.

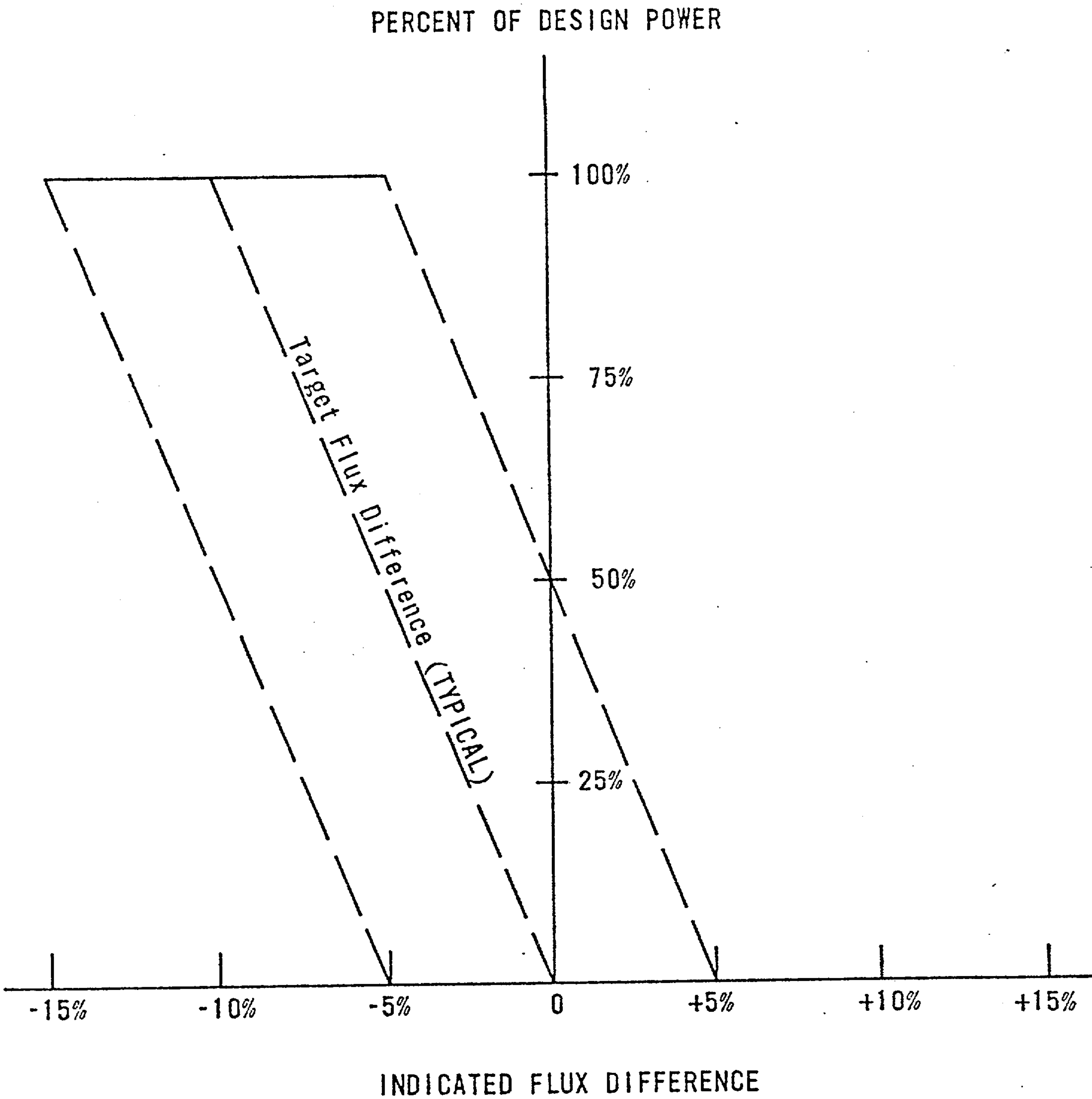
As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This can be accomplished without part length rods by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Operating Transient events, the core is protected from overpower and a minimum DNBR of less than 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Operating Transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

A quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the core deisgn radial power distribution. A 2% tilt alarm set point represents a minimum practical value consistent with instrumentation error and operating procedures. This asymmetry level is sufficient to detect significant misalignment of control rods which is the most likely cause of radial power asymmetry.

REFERENCES

FSAR - Section 14.3.2



21

Figure B 3.2-1 Target Band on Indicated Flux Difference as a Function of Operating Power Level (Typical for BOL)

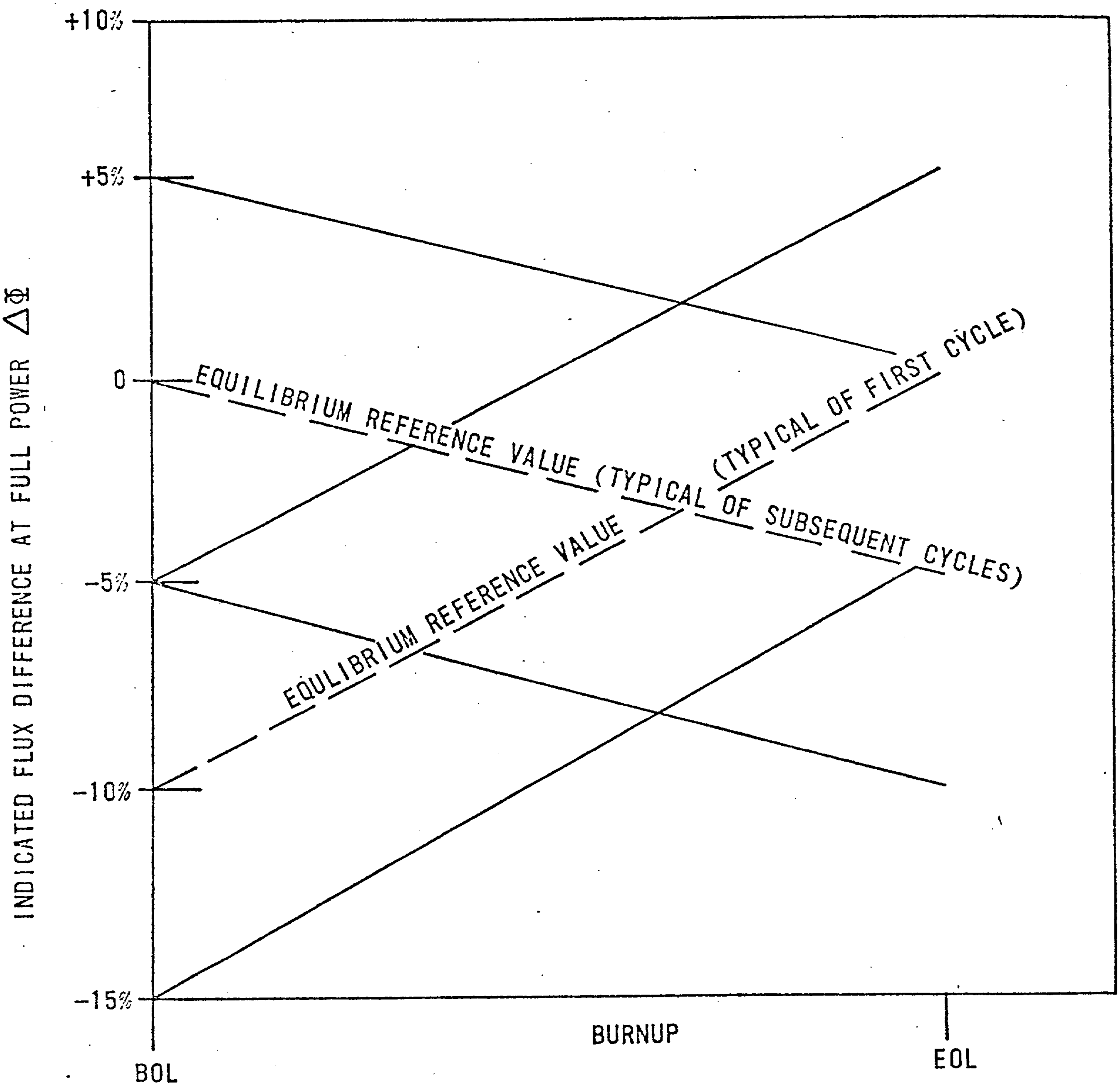


Figure B 3.2-2 Permissible Operating Band on Indicated Flux Difference as a Function of Burnup (Typical)

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGE TO THE

APPENDIX A TECHNICAL SPECIFICATIONS OF LICENSE NOS. DPR-31 AND 41

TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4

DOCKET NOS. 50-250 & 50-251

The U. S. Nuclear Regulatory Commission (the Commission) has reviewed the licensee's proposed change to the Appendix A Technical Specifications of Facility Operating Licenses DPR-31 and DPR-41. This change would authorize the Florida Power and Light Company to operate the Turkey Point Nuclear Generating Units 3 and 4 with certain revisions to the present limiting conditions for operation specified in Appendix A of the referenced licenses. These revisions result from the implementation of the Acceptance Criteria For the Emergency Core Cooling Systems for Light Water Nuclear Power Reactors (ECCS) as specified in Section 50.46 of 10 CFR Part 50. In conjunction with these changes additional revisions are requested which are associated with the Turkey Point Unit 4. No revisions to the Environmental Technical Specifications, (Appendix B) have been requested as a result of this proposed change.

The Commission's Division of Reactor Licensing has prepared an environmental impact appraisal for the proposed change to the Appendix A Technical Specifications, for Facility Operating Licenses DPR-31 and DPR-41.

On the basis of the environmental impact appraisal presented in this document, we have concluded that an environmental impact statement for this particular action is not warranted because, pursuant to the Commission's regulations in 10 CFR 51 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6, the Commission has determined that this proposed change to the Appendix A Technical Specifications is not a major federal action significantly affecting the quality of the human environment. The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. 20555, and at the Environmental and Urban Affairs Library, Florida International University, Miami, Florida 33199.

Dated at Rockville, Maryland, this 5th day of June, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Chief
Environmental Projects Branch No. 1
Division of Reactor Licensing

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING: AMENDMENT NO. 9 TO LICENSE NO. DPR-31

AMENDMENT NO. 8 TO LICENSE NO. DPR-41

CHANGE NO. 21 TO THE TECHNICAL SPECIFICATIONS (APPENDIX A)

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4

I. Description of Proposed Action

By letters dated February 7, 10, and 13, 1975, March 10, 11, and 15, 1975, April 30, 1975, and May 13, 15, and 21, 1975, the Florida Power and Light Company(the licensee) provided information and supportive analysis relative to a proposed change in Appendix A Technical Specifications of Facility License Nos. DPR-31 and DPR-41. The proposed change concerns (a) revisions to the limiting conditions for operation to the Turkey Point Nuclear Generating Units 3 and 4 as a result of the implementation of the Acceptance Criteria for the Emergency Core Cooling System (ECCS) and (b) Fuel Cycle 2 operation of Turkey Point Unit 4 at 2250 psia and a maximum of 30,000 effective full power hours (EFPH).

II. Environmental Impacts of Proposed Action.

A Final Environmental Statement (FES) was issued in July 1972 which concluded that based upon an evaluation of the proposed operating conditions, an operating license should be issued for Turkey Point Unit 3, and to Unit 4 when completed. The licensee in their letter of April 30, 1975, has indicated that there would be no environmental impact associated with the proposed license amendments. The NRC conducted an environmental impact appraisal of these proposed license amendments as required by the NEPA and Section 51.7 of 10 CFR Part 51. The potential NEPA concerns associated with both the implementation of the ECCS Criteria for Units 3 and 4 and operation of Unit 4 subsequent to Cycle 2 fuel reload can be defined as:

1. Changes in benefits accruing from plant operation due to revisions to reactor power limits.
2. Variation in environmental impacts resulting from changes in non-radiological effluent releases.
3. Variation in environmental impacts resulting from changes in radiological effluent releases.

A. <u>ECCS Criteria for Turkey Point Units 3 and 4</u>

The proposed change to incorporate the ECCS Acceptance Criteria in Units 3 and 4 has been evaluated in terms of these potential NEPA concerns. This evaluation has confirmed that operating power will remain as previously evaluated. As such, no resultant effects upon the Cost/Benefit balance developed for the Turkey Point Units and presented in the FES of July 1972 are anticipated.

Since this change will not result in modified power levels no environmental impacts (other than expressed in the FES) resulting from radiological and non-radiological effluents are predicted. The FES evaluation of the Turkey Point Plant's cooling water flow, thermal effluents, chemical effluents, radiological source term and effluents during normal operation and post-accident conditions need not be revised as a result of the implementation of the ECCS acceptance criteria.

B. <u>Turkey Point Unit 4 - Revisions to Operating Parameters</u>

The operation of Unit 4 with certain revisions to operating parameters following Cycle 2 core reload has been evaluated with respect to the identified NEPA areas of concern. The fuel shuffling for Cycle 2 will not result in any unreviewed environmental impacts. The revision in operating pressure, resulting in an increase from 1900 to 2250 psia is associated with fuel assembly integrity and operating margins and will have no detectable impacts on the environment not previously evaluated. The revision of the fuel EFPH limit from 24,500 to 30,000 is associated with fuel element integrity and does not involve a power level or fuel utilization change. This modification will not result in any environmental impacts in excess of those previously evaluated by the staff.

III. Conclusions and Basis for Negative Declaration

On the basis of the NRC evaluation presented above and information supplied by the licensee, it is concluded that both the implementation of the ECCS Acceptance Criteria for Turkey Point Units 3 and 4 and the operation of Unit 4 following Cycle 2 fuel reload will produce no discernible environmental impacts other than those previously addressed in the FES of July 1972. It is not anticipated that the issuance of this change to the Appendix A Technical Specifications will affect the Cost/Benefit balance nor the evaluation of the radiological and non-radiological effluents presented in the FES, and it will not require changes to the Environmental Technical Specifications (Appendix B).

Having reached these conclusions, the Commission has determined that an environmental impact statement need not be prepared for the proposed license amendments and that a Negative Declaration shall be issued to this effect.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 TO LICENSE NO. DPR-31, AND

AMENDMENT NO. 8 TO LICENSE NO. DPR-41

(CHANGE NO. 21 TO TECHNICAL SPECIFICATIONS)

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4

DOCKET NOS. 50-250 AND 50-251

INTRODUCTION

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR §50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading, "the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, §50.46". The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendment as may be necessary to implement the evaluation results.

Florida Power and Light Company (FPL) has requested a license amendment which will allow Turkey Point Unit 4 operation following reload for core cycle 2. This license amendment request included analyses of the applicability of previously performed safety analyses and proposed Technical Specification changes based on the Unit 4 core configuration for cycle 2.

As required by our Order of December 27, 1974, FPL has also submitted an ECCS reevaluation and related Technical Specifications. The ECCS reevalution applies also to Turkey Point Unit 3 which initiated core cycle 2 operation in the fall of 1974. Since there are no significant differences between the core configurations for Unit 4 cycle 2 and Unit 3 cycle 2, the ECCS reevaluation and related Technical Specifications apply to both Units 3 and 4.

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The first part of this safety evaluation, "Unit 4 Core Cycle 2 Reload", discusses and evaluates the requested action regarding the Turkey Point Unit 4 core cycle 2 reload. The second part of this safety evaluation, "Emergency Core Cooling System", discusses and evaluates the ECCS reevaluation and related Technical Specifications which are applicable to both Units 3 and 4.

PART I: UNIT 4 CORE CYCLE 2 RELOAD

A. Introduction

By letter dated February 10, 1975, Florida Power and Light Company (FPL) proposed changes to the Technical Specifications of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Nuclear Generating Units 3 and 4. Supplemental information relating to the requested changes was supplied by FPL in their letters of February 7, and 13 and April 30 and May 13, 1975. FPL requested: (1) the Unit 4 allowable fuel residence time (the minimum predicted time to clad flattening) be increased from 24,500 effective full power hours (EFPH) to 30,000 EFPH and (2) the overtemperature ΔT, overpower ΔT and pressurizer low pressure limiting safety system settings be modified to allow Unit 4 operation at 2250 psia following refueling, and (3) the Unit 4 control rod insertion limits for two loop operation, a limiting condition for operation (LCO), be modified.

Because Units 3 and 4 share joint Technical Specifications, FPL proposed modifying the Technical Specifications for Unit 3 to reflect the proposed revision to the Unit 4 Technical Specifications. However, the operating limits for Unit 3 are unchanged by the Unit 4 reload for core cycle 2.

B. Discussion

1. Reactor Core Description

The Unit 4 core loading for fuel cycle 2 will include 56 prepressurized fuel assemblies. These assemblies, which are known as Region 4 fuel assemblies, have a slightly higher enrichment than do the Region 1 fuel assemblies they replace (2.55% U-235 vs. 1.85% U-235). However, the enrichment of the Region 4 fuel assemblies are within the enrichment range of other assemblies presently installed in the Unit 4 reactor (1.85% U-235 to 3.11% U-235). The increased enrichment compensates for the fission product reactivity poisoning produced within the reactor during core cycle 1 operation. The Region 4 fuel assemblies have been fabricated by Westinghouse Electric Corporation, the fabricator of the fuel assemblies now loaded in the Turkey Point Units. They are mechanically identical to the presently installed fuel assemblies with the exception of two assemblies which have provisions for removable fuel rods.

The Unit 4 cycle 2 core configuration was intended to be identical, with regard to fuel enrichment, to the Unit 3 cycle 2 core. Unit 3 has been operating in cycle 2 since November 1974. However, a Region 3 fuel assembly was damaged during the Unit 4 refueling necessitating the substitution of four additional Region 4 fuel assemblies for four Region 3 fuel assemblies. The desire to maintain core symmetry in the four core quadrants necessitated the substitution of four instead of one fuel assembly.

2. Minimum Time to Clad Flattening

Turkey Point Units 3 and 4 have been operating at a reduced primary pressure of 1900 psia since early in core cycle 1. Reduced primary pressure was initiated during cycle 1 in order to lengthen the predicted time to clad flattening by reducing the pressure differential across the fuel cladding and thus reducing the clad creep rate. The presently specified Unit 4 fuel residence limit of 24,500 EFPH is the analytically determined minimum time to clad flattening for Unit 4 Region 3 fuel assemblies using a previously approved model and assuming continued reactor operation at 1900 psia.

Westinghouse has recently revised the clad collapse model and has submitted reports WCAP-8377(1) and WCAP-8381(2) which described the revised model. The revised model as described in the referenced reports has been approved for licensing actions and was used in support of Turkey Point Unit 3 License Amendment No. 6(3). The revised model as described in License Amendment No. 6 predicts longer times to clad flattening. Since the predicted time to clad flattening for Unit 4 now exceeds the expected life of the Unit 4 fuel assemblies, there is no longer an advantage for operation at reduced pressure. Therefore, FPL has stated that they plan to return Unit 4 to 2250 psia primary system pressure following reload for core cycle 2.

3. Overtemperature ΔT and Overpower ΔT

The core protection system operates by defining a region of permissible operation in terms of power, pressure, temperature, coolant flow and axial power distribution. This allowable operating region with regard to coolant temperature difference across the reactor core is determined by the equations which define the overtemperature ΔT and overpower ΔT reactor trips. The overtemperature ΔT reactor trip protects the core against nucleate boiling, excessive hot channel exit quality, and hot channel boiling for any combination of power, pressure, temperature, and axial core power distribution. Similarly, the overpower ΔT reactor trip provides protection against exceeding fuel rod design limits for accidents involving overpower excursions.

FPL, in order to resume reactor operation at 2250 psia, has proposed modifying the overtemperature ΔT and overpower ΔT reactor trip expressions and has proposed a new pressurizer low pressure trip setpoint which is consistent with the new overtemperature ΔT and overpower ΔT trip setpoint expressions.

4. Control Rod Insertion Limits

Control of the operating reactor is provided by neutron absorbing control rods and soluble boric acid in the reactor coolant. The more boric acid contained in the reactor coolant the less the control rods need to be inserted to provide reactor control. The specified control rod insertion limits are the result of analyses performed for the Unit 4 cycle 2 core configuration to insure: (1) an adequate shutdown margin is maintained throughout cycle life, (2) hot channel factors are maintained below design limits, (3) acceptable consequences of a rod ejection accident, and (4) acceptable consequences of rod misalignment. The maintenance of adequate shutdown margin at the end of core life is the consideration which typically defines the control rod insertion limits.

The proposed control rod insertion limits for Unit 4 are identical to those now in effect for Turkey Point Unit 3 which were previously approved prior to the initiation of Unit 3 core cycle 2 operation.

C. Evaluation

1. Reactor Core Description

FPL's analysis of the loading pattern and their comparison of core physics parameters for core cycle 2 with those of core cycle 1 indicates the nuclear parameters for core cycle 2 fall within the range of values assumed in the Turkey Point Final Safety Analysis Report (FSAR). We agree that this comparison shows there are no significant differences between the Unit 4 cycle 1 and cycle 2 core configurations. Therefore, the consequences of previously analyzed accidents and transients are not increased by the Unit 4 reload for core cycle 2 and since these consequences were previously determined to be acceptable for Turkey Point, the conclusions of previous safety evaluations are unchanged by the core reload.

2. Minimum Time to Clad Flattening

The presently specified Unit 4 fuel residence limit of 24,500 EFPH is the analytically determined minimum time to clad flattening for Unit 4 Region 3 fuel assemblies using a previously approved model and assuming continued reactor operation at 1900 psia. The minimum time to clad flattening for fuel regions other than Region 3 exceeds the time for Region 3 fuel assemblies because of higher initial fuel rod internal pressure in these other assemblies. Region 3 fuel assemblies have the most limiting time to clad flattening regardless of the model used to predict time to clad flattening.

FPL has recalculated the minimum time to clad flattening using the approved model described in WCAP-8377 and WCAP-8381. FPL has determined this time to be 30,000 EFPH for Unit 4 cycle 2 fuel assemblies, assuming reactor operation at 2250 psia. The results and conclusions of previous safety evaluations and previously approved operating limits, now in effect, remain unchanged as long as clad flattening is predicted not to occur. We have reviewed FPL's request and have approved the requested Unit 4 fuel residence time. Our approval is based on FPL's use of the approved revised clad flattening model and our independent determination that the model was used to determine the minimum time to clad flattening for the most critical assemblies in the cycle 2 fuel loading.

3. Overtemperature ΔT and Overpower ΔT

FPL plans to resume Unit 4 reactor operation at 2250 psia following refueling for core cycle 2. FPL has, therefore, proposed overtemperature ΔT, overpower ΔT and pressurizer low pressure reactor trip settings to be consistent with 2250 psia operation. In proposing the new trip setpoints, FPL also proposed additional margin in the trip setpoints in order to accommodate setpoint tolerance. These margins were not allowed at this time as FPL did not provide sufficient justification for their inclusion in the setpoint values and expressions.

We have approved use of overtemperature ΔT, overpower ΔT and pressurizer low pressure trip setpoint expressions and values that were in effect prior to the initiation of 1900 psia reactor operation. Since we have concluded that there are no significant differences between the cycle 1 and cycle 2 core loading, these previously approved and used reactor trip setpoints will provide adequate protection against fuel clad damage in the event of any postulated operating transient.

4. Control Rod Insertion Limits

FPL has analyzed the control rod insertion limits for both three loop and two loop operation and has proposed Unit 4 control rod insertion limits for two loop operation which are more conservative than those presently specified. FPL has not proposed a change to the control rod insertion limits for three loop operation as their analysis shows the specified three loop insertion limits are more conservative than necessary. We have reviewed FPL's request and since we find that the use of the proposed presently specified three loop control rod insertion limits and the proposed two loop control rod insertion limits will not effect previously performed applicable safety analyses, we approve the proposed insertion limits. Use of the more conservative proposed control rod insertion limits will increase the minimum available shutdown margin, maintain an acceptable core power distribution, decrease the consequences of a control rod ejection accident and decrease the consequences of control rod misalignment.

D. Summary

Our evaluation supports the conclusions that: (1) clad flattening is predicted not to occur during a projected fuel residence time of 30,000 EFPH, (2) the use of reactor trip setpoints previously approved for 2250 psia operation are appropriate for use during core cycle 2 operation at 2250 psia and (3) the proposed control rod insertion limits are conservative when compared to those now in effect. We have determined that there are no significant differences between core cycle 1 and core cycle 2 and that the conclusions of our earlier safety evaluations remain unchanged for both 1900 psia and 2250 psia operation of clad flattening does not occur. We further conclude that operation using the proposed Technical Specifications will not endanger the health and safety of the public and will be in compliance with the Commission's regulations.

PART II: EMERGENCY CORE COOLING SYSTEM

A. Introduction

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling System for Light

Water Nuclear Power Reactors"[4]. One of the requirements of the Order was that the licensee shall submit a reevaluation of the ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, § 50.46. The Order also required that the evaluation shall be accompanied by such proposed changes in the Technical Specifications or license amendment as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, Florida Power and Light Company (FPL) has submitted an ECCS reevaluation and related Technical Specifications. The reevaluation and Technical Specifications, which are applicable to the core configuration for Turkey Point Nuclear Generating Units 3 and 4 core cycle 2, were submitted by letters dated March 10 and 11, 1975.

During the course of our review of FPL submittals, FPL supplied supplemental information in their letters of April 10 and 30, 1975 and May 15 and 21, 1975. Proposed Technical Specification changes regarding the safety injection system accumulator water volume were proposed by FPL in their letter of September 27, 1974. Notice of the proposed action was published in the Federal Register on April 9, 1975 (40 F.R. 16152).

B. Discussion

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report (SER) of the Turkey Point Units 3 and 4 dated December 27, 1974.

The background of the staff review of the Westinghouse ECCS models and their application to Turkey Point is described in the staff SER for this facility dated December 27, 1974 (the December 27, 1974, SER) issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 and the November 1974 Supplement to the Status Report which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also described the various changes required in the earlier Westinghouse evaluation model. Together, the December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model.

The Turkey Point ECCS evaluation which is covered by this safety evaluation properly conforms to the accepted model. The March 10, 1975 submittal contained: (1) analyses of sufficient break sizes and location to verify that the worst break condition had been considered and (2) documentation, by reference to submitted Westinghouse Topical Reports, of the ECCS model modifications described in our December 27, 1974 SER. However, the March 10, 1975 submittal did not contain an ECCS evaluation for reactor operation with less than three reactor coolant pumps in operation. Therefore, the Technical Specifications have been modified to prevent reactor operation with less than three operating reactor coolant pumps.

Subsequent to their submittal of March 10, 1975, FPL reviewed the ECCS capabilities and operating procedures to determine that system capabilities and operating procedures assure that boron precipitation will not compromise core cooling capability. These system capabilities and operating procedures were described by FPL in their letters dated April 30 and May 21, 1975.

C. Evaluation

We have reviewed the evaluation of ECCS performance submitted by FPL for the Turkey Point Nuclear Generating Units 3 and 4 and concluded that the evaluation has been performed wholly in conformance with the requirements of Appendix K to 10 CFR Part 50. Therefore, operation of the reactor would meet the requirements of 10 CFR § 50.46 provided that the reactor is operated in accordance with the proposed Technical Specifications as modified by subsequent NRC review.

The analyses submitted on March 10, 1975, identified the worst break as the double-ended cold leg guillotine break with a discharge coefficient of 0.4. The calculated peak clad temperature for this break is 2150°F, a value sufficiently below the acceptable limit of 2200°F as specified in 10 CFR § 50.46(b). In addition, the calculated maximum local metal water reaction of 7.4% and total core wide metal/water reaction of less than 0.3% are well below the allowable limits of 17% and 1%, respectively.

The ECCS reevaluation was performed assuming a reactor operating pressure of 2250 psia. Since Unit 3 is operating at 1900 psia and Unit 4 will operate at 2250 psia following refueling, FPL performed an analysis of the effect of reduced pressure operation on computed ECCS performance(5). This evaluation showed that the effect of operating at 1900 psia would theoretically reduce the maximum peak clad temperature following a LOCA. In actual operation, the blowdown phase from 2250 psia to 1900 psia lasts less than 100 milliseconds and has a negligible effect on peak clad temperature. Use of a higher primary system pressure in ECCS analysis is conservative with respect to resulting peak clad temperature, clad oxidation, and hydrogen generation and conforms to the requirements of 10 CFR § 50.46.

The ECCS containment pressure calculations for Turkey Point were done using the Westinghouse ECCS evaluation model. We reviewed Westinghouse's model and published a Status Report on October 15, 1974(6), which was amended November 13, 1974(7). We concluded that Westinghouse's containment pressure model was acceptable for ECCS evaluation and required that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant.

This information was submitted for Turkey Point by letter dated December 6, 1974. FPL has reevaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for ECCS analysis. This evaluation was based on measurements within the containment and from as-built drawings to which additional margin was added. The containment heat removal systems were assumed to operate at their maximum capacities, and minimum operational values for the spray water and service water temperatures were assumed.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Turkey Point is conservative and therefore the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

To assure that peak linear heat generation rates consistent with the ECCS reevaluation are not exceeded, the Technical Specifications include: (1) hot channel factor limits, (2) axial flux difference limits, and (3) quadrant power tilt limits. These specified limits on power distribution maintain operating conditions for both Unit 3 and Unit 4 consistent with the assumptions used in the ECCS reevaluation.

We have also reviewed FPL's submittals regarding the effect of: (1) single component failure on ECCS operation and (2) boron precipitation within the reactor vessel on long term cooling.

A review of the Turkey Point piping and instrumentation diagrams indicated that actuation of specific motor operated valves resulting from spurious failures in the valve control circuits could affect ECCS operation. We identified the following motor operated valves as not satisfying the single failure criterion because a failure in the valve operator control circuit could move the valve to a position which could affect ECCS operation.

Valve Number	Location
864A, 864B	Refueling Water Storage Tank Discharge
862A, 862B	Residual Heat Removal Pump Suction Line
865A, 865B, 865C	Accumulator Discharge
866A, 866B	Safety Injection Pump Discharge to Hot Legs
758	Control Valve Residual Heat Exchanger Discharge

The Technical Specifications have been modified so that: (1) power is removed from the valve motor operators by locking open the circuit breakers at their motor control center or (2) in the case of valve No. 758, the air supply to the valve operator is shut off. Thus, no single failure of any of these valves will adversely affect ECCS operation.

FPL submitted by letter dated April 30, 1975, emergency operating procedures for the long term post-LOCA core cooling period. The procedures are intended to prevent excessive concentration of the boron in the reactor vessel. The Turkey Point operating procedures were supported by a Westinghouse generic analysis[8]. We have reviewed the Turkey Point procedures and the referenced analysis and have concluded that the ECCS can be operated in a manner that will prevent excessive boron concentration within the reactor core. During our review of the Turkey Point long term cooling procedures, we suggested that the initial cold leg injection period be modified from the originally proposed 24 hours to 20 hours. We also suggested that following the initial 20 hour cold leg injection period, FPL should: (1) inject diluent simultaneously through the hot and cold legs or (2) alternate diluent injection through the hot and cold legs, with sufficiently short periods between the hot and cold leg injection periods to prevent excessive boron buildup within the reactor core.

In their letter of May 21, 1975, FPL modified their original procedures to include our recommendations. In addition, FPL also modified their original long term cooling procedures to specify that motor-operated valves 866A and 866B in the safety injection (SI) pump discharge line to the hot legs and motor-operated valves 750 and 751 in the hot leg suction line be opened within 2 hours following a LOCA even though hot leg injection will not be required for 20 hours. FPL has changed their procedures so that these valves can be opened within 2 hours because the valve motor operators have been proven by test to be operable only for periods up to 12 hours in a post-LOCA environment. Valves 866A and 866B are locked in the closed position during normal operation so that a safety injection pump initiation signal will not initiate hot leg injection. Hot leg injection is only used to recirculate reactor coolant water from the containment during the post-LOCA recirculation phase. The recirculation phase occurs hours after the LOCA and ample time is available to unlock the circuit breakers. Prior to opening valves 866A, 866B, 750 and 751, following a LOCA, valve 869 in the hot leg injection header and valve 741A in the RHR supply line will be closed. Closing valves 869 and 741A will prevent hot leg injection and since these valves are located outside containment the valve position can be visually verified. Long term cooling using hot leg injection will then be controlled using valves 869 and 741A.

In our review of the proposed procedures, we also identified a pipe in the hot leg injection system whose rupture could prevent hot leg injection. In the event that this pipe ruptures, during or following a LOCA, FPL has developed an alternate procedure for instituting hot leg injection. Using this alternate procedure, water would be pumped by the reactor heat removal (RHR) pumps from the containment sump through the hot leg suction line normally used for shutdown cooling into the reactor hot leg piping. We have concluded that systems exist and procedures have been adopted by FPL which will prevent excessive boron concentration in the reactor vessel during the long term cooling period in the highly unlikely event that such actions are required.

D. Summary

Our evaluation supports the conclusion that: (1) the evaluation has been performed wholly in conformance with the requirements of Appendix K to 10 CFR § 50.46 and (2) ECCS cooling performance for the Turkey Point Nuclear Generating Units 3 and 4 will conform to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR 50.46(b). In addition we have concluded that (1) ECCS cooling performance will be adequate despite any postulated failure of any single component and (2) that adequate systems and procedures exist to provide reasonable assurance that boron precipitation will not occur within the reactor vessel.

PART III: CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: JUN 5 1975

BIBLIOGRAPHY

(1) George, R. A., Lee, Y. C., and Eng, G. H., "Revised Clad Flattening Model," Westinghouse Electric Corporation, WCAP-8377, (Proprietary) July 1974.

(2) George, R. A., Lee, Y. C., and Eng, G. H., "Revised Clad Flattening Model," Westinghouse Electric Corporation, WCAP-8381, July 1974.

(3) Amendment No. 6 to Facility Operating License No. DPR-31 and Amendment No. 5 to Facility Operating License No. DPR-41 for Turkey Point Nuclear Generating Units 3 and 4 (Docket Nos. 50-250 and 50-251) including Technical Specification Change No. 18, February 19, 1975.

(4) "Order for Modification of License", letter to Florida Power and Light Company from George Lear, December 27, 1974.

(5) FSAR Supplement No. 59 (Revision 35), page 14.3.2-1, September 6, 1974.

(6) "Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K", October 15, 1974.

(7) "Supplement to the Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K", November 13, 1974.

(8) CLC-NS-309, Letter to T. M. Novak from C. L. Caso, Westinghouse Nuclear Energy Systems, April 1, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET 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 Amendment Nos. 9 and 8, respectively, to Facility Operating Licenses Nos. DPR-31 and DPR-41 issued to Florida Power and Light Company which revised Technical Specifications for operation of Turkey Point Nuclear Generating Units 3 and 4, located in Dade County, Florida. These amendments are effective as of date of issuance.

These amendments: (1) incorporate operating limits in the Technical Specifications for the facilities based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR 50, and (2) modify certain Unit 4 operating limits to reflect the results of the cycle 2 core performance analysis.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notices of Proposed Issuance of Amendments to Facility Operating Licenses in connection with these actions were published in the FEDERAL REGISTER on March 20, 1975 (40 F.R. 12717) and on April 9, 1975 (40 F.R. 16152). No request for a hearing or petition for leave to intervene was filed following notice of the proposed actions.

For further details with respect to these actions, see (1) the applications for amendments dated September 27, 1974, February 10, and March 11, 1975, and Supplements dated February 7 and 13, March 10, April 10 and 30, and May 13, 15, and 21, 1975, (2) Amendment No. 9 to License No. DPR-31 and Amendment No. 8 to License No. DPR-41 with Change No. 21, (3) the Commission's related Safety Evaluation, and (4) the Commission's Negative Declaration dated (which is also being published in the FEDERAL REGISTER) and associated Environmental Impact Appraisal. 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), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 5th day of June, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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