NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-250

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TO: Mr. V. Stello

FROM: FPL
Miami, Fla. 33101
R.E. Uhrig

DATE OF DOCUMENT 12-21-76

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DESCRIPTION: Ltr notarized 12-21-76 requesing for Amdt to OL/Tech Specs....trans the following:

PLANT NAME: Turkey Pt. Unit 3

ENCLOSURE: Revised & addl pages to OL/Tech Specs pertaining to ECCS cooling performance....

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ACKNOWLEDGED

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FOR ACTION/INFORMATION

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ECCS

NRC FORM 195 (2-76)

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P. O. BOX 013100, MIAMI, FL 33101
FLORIDA POWER & LIGHT COMPANY

December 21, 1976
L-76-429

Office of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

Re: Turkey Point Unit 3
Docket No. 50-250
Proposed Amendment to Facility
Operating License DPR-31

In accordance with 10 CFR 50.30 Florida Power & Light Company hereby submits three (3) signed originals and forty (40) copies of a request to amend Appendix A of Facility Operating License DPR-31.

This proposal is being submitted as a result of a re-evaluation of ECCS cooling performance calculated in accordance with an approved Westinghouse Evaluation Model. The ECCS re-evaluation was forwarded to you on December 9, 1976 under our cover letter L-76-419. The proposed changes are described below and shown on the accompanying Technical Specification pages bearing the date of this letter in the lower right hand corner. NRC approval of the proposed amendment is requested prior to the end of the current refueling outage, however, the amendment is not necessary for the conduct of the refueling, or return to operation following completion of refueling.

Page 3.2-3

Page 3.2-3 is designated applicable to Unit 3 only. The page contains a revision to Specification 3.2.6.a such that the limit on Heat Flux Hot Channel Factor (F_q) for Unit 3 is reduced from 2.32 to 2.24.

Page 3.2-3a

Page 3.2-3a is designated applicable to Unit 4 only.

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Pages B3.2-4 and B3.2-6

Pages B3.2-4 and B3.2-6 are designated applicable to Unit 3 only. These pages present the basis for the revised Unit 3 F_q limit.

Pages B3.2-4a and B3.2-6a

Pages B3.2-4a and B3.2-6a are designated applicable to Unit 4 only.

Page 3.4-1

Page 3.4-1 is designated applicable to Unit 3 only. The accumulator water volume in Specification 3.4.1.3 is revised from 825-841 ft^3 to 875-891 ft^3.

Page 3.4-1a

Page 3.4-1a is designated applicable to Unit 4 only.

The proposed amendment has been reviewed by the Turkey Point Plant Nuclear Safety Committee and the Florida Power & Light Company Nuclear Review Board. They have concluded that it does not involve an unreviewed safety question. A safety evaluation is attached.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU/MAS/cpc

Attachments

cc: Mr. Norman C. Moseley
Robert Lowenstein, Esquire

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

Robert E. Uhrig, being first duly sworn, deposes and says:

That he is a Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

Robert E. Uhrig
Robert E. Uhrig

Subscribed and sworn to before me this
21 day of December, 1976

Theresa Garceau
NOTARY PUBLIC, in and for the County of Dade,
State of Florida

My commission expires: NOTARY PUBLIC STATE OF FLORIDA AT LARGE
MY COMMISSION EXPIRES JAN. 26, 1979
BONDED THRU GENERAL INSURANCE UNDERWRITERS

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:

$F_q(Z) \leq (2.24/P) \times K(Z)$ 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 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,

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6. <u>POWER DISTRIBUTION LIMITS</u>

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

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

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

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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 2.24 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

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 2.24 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-

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SAFETY EVALUATION

1.0 Introduction

This safety evaluation supports the following proposed changes to the Unit 3 Technical Specifications:

The maximum allowable nuclear peaking factor (F_q) is decreased from 2.32 to 2.24.

The limits on Safety Injection accumulator water volume are increased from 825-841 ft^3 to 875-891 ft^3.

2.0 Discussion

ECCS Re-evaluation

A re-evaluation of ECCS cooling performance calculated in accordance with an approved Westinghouse Evaluation Model has been performed. The re-evaluation shows that for breaks up to and including the double ended severence of a reactor coolant pipe, the ECCS will meet the Acceptance Criteria presented in 10 CFR 50.46. The detailed re-evaluation is contained in FPL letter L-76-419 of December 9, 1976, and shows that, at a core power level of 102% of 2192 Mwt and a minimum accumulator water volume of 875 ft^3 per accumulator, the maximum allowable nuclear peaking factor is 2.25. However, since the Technical Specifications allow a maximum core power level of 2200 Mwt, the re-evaluation is being revised using the higher power level. The revised calculation is expected to yield a maximum F_q of 2.24.

3.0 Conclusions

Based on these considerations, (1) the proposed change does not increase the probability or consequences of accidents or malfunctions of equipment important to safety and does not reduce the margin of safety as defined in the basis for any technical specification, therefore, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.