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MAR 17 1982

Docket Nos. 50-250
and 50-251

Posted
Amdt. 74
to DPR-41
(See Correction
letter of
5-26-82)

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

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 80 to Facility Operating License No. DPR-31 and Amendment No. 74 to Facility Operating License No. DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated May 14, 1981, as supplemented November 23, 1981 and January 28, 1982.

These amendments revise the Technical Specifications to specify new power distribution limits for base load and radial burndown operation.

We have received you letter dated March 15, 1982 which satisfies the current requirements of Technical Specification 6.9.3 for Unit 3. Such a report is not necessary for Unit 4 at this time because P_T is not less than 1.

We have found it necessary to make changes in certain of the Technical Specifications. We have discussed the changes with your staff. They found the changes acceptable and the changes have been incorporated.

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

Sincerely,

ORIGINAL SIGNED

Marshall Grotenhuis, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 80 to DPR-31
2. Amendment No. 74 to DPR-41
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

*Previous concurrence see next page

OFFICE	ORB#1:DL*	ORB#1:DL*	ORB#1:DL*	AD/OR:DL*	OELD*		
SURNAME	CParrish	MGrotenhuis	SVarga	TNovak			
DATE	03/ /82	03/16/82:ds	03/ /82	03/ /82	03/ /82		

Robert E. Uhrig
Florida Power and Light Company

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
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 May 14, 1981, as supplemented November 23, 1981 and January 28, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-31 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 80 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 17, 1982

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. DPR-41

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated May 14, 1981, as supplemented November 23, 1981 and January 28, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-41 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 74 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 17, 1982

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.2-2	3.2-3
-----	3.2-3a
-----	3.2-3b
-----	3.2-3c
3.2-4	3.2-4
Figure 3.2-3	Figure 3.2-3
Table 4.1-1 (sheet 1)	Table 4.1-1 (sheet 1)
6-22	6-22
B3.2-3	B3.2-3
B3.2-4	B3.2-4
B3.2-5	B3.2-5
B3.2-6	B3.2-6
B3.2-7	B3.2-7
B3.2-8	B3.2-8
------	B3.2-8a

reactivity insertion upon ejection greater than 0.3% $\Delta k/k$ 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 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 is not operable, rod positions shall be logged once per shift and after a load change greater than 10% of rated power. If both alarms are inoperable for two hours or more, the nuclear overpower trip shall be reset to 93% of rated power.

6. POWER DISTRIBUTION LIMITS

a. Hot channel factors:

(1) F_Q Limit

The hot channel factors (defined in Bases) must meet the following limits at all times except during low power physics tests:

$$F_Q(Z) \leq ([F_Q]_L/P) \times K(Z), \text{ for } P > 0.5$$

$$F_Q(Z) \leq (2 \times [F_Q]_L) \times K(Z), \text{ for } P \leq 0.5$$

$$F^N_{\Delta H} \leq 1.55\,[1.0 + 0.2\,(1 - P)]$$

Where P is the fraction of rated power at which the core is operating; K(Z) is the function given in Figure 3.2-3; Z is the core height location of F_Q. $[F_Q]_L$ and K(Z) are dependent on the steam generator tube plugging level as follows:

Plugging level	$[F_Q]_L$	Figure Number for K(Z)
$\leq$ 28%	2.125	3.2-3

(2) Augmented Surveillance (MIDS)

If $[F_Q]_P$, as predicted by approved physics calculations, exceeds $[F_Q]_L$ then the power will be limited to a turnon power fraction, P_T, equal to the ratio of $[F_Q]_L$ divided by $[F_Q]_P$, or, for operation at power levels above P_T, augmented surveillance of hot channel factors shall be implemented, except in Base Load operation (Section 3.2.6.a(3)) or Radial Burndown operation (Section 3.2.6.a(4)). If $[F_Q]_P$, as predicted by approved physics calculations, is less than $[F_Q]_L$ (i.e.: $P_T \geq 1.00$), operation in accordance with Augmented Surveillance (MIDS)(Sections 3.2.6.a(2)) Baseload Operation (Section 3.2.6.a(3)) or Radial Burndown Operation (Section 3.2.6.a(4)) is not required.

For operation at power levels between P_T and 1.00, the following shall apply when not in baseload or radial burndown operation:

1. The axial power distribution shall be measured by MIDS when the thermal power is in excess of P_T such that the limit of $[F_Q]^L/P$ times Figure 3.2-3 is not exceeded. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation (Z).

   (1) If $F_j(Z)$ exceeds $[F_j(Z)]_S$ as defined in the bases by $\leq$ 4%, immediately reduce thermal power one percent for every percent by which $[F_j(Z)]_S$ is exceeded.

   (2) If $F_j(Z)$ exceeds $[F_j(Z)]_S$ by > 4% immediately reduce thermal power below P_T. Corrective action to reduce $F_j(Z)$ below the limit will permit return to thermal power not to exceed current P_L as defined in the bases.

2. $F_j(Z)$ shall be determined to be within limits by using MIDS to monitor the thimbles required per specification 6.a.2-3 below at the following frequencies:

   (1) At least once every 24 hours, and

   (2) Immediately following and as a minimum at 2, 4 and 8 hours following the events listed below and every 24 hours thereafter

      1) Raising the thermal power above P_T, or
      2) Movement of control-bank D more than an accumulated total of 15 steps in any one direction.

3. MIDS shall be operable when the thermal power exceeds P_T with:

   (1) At least two thimbles available for which R_j and $_j$ as defined in the bases have been determined.

   (2) At least two movable detectors available for mapping F_j (Z).

   (3) The continued accuracy and representativeness of the selected thimbles shall be verified by using the most recent flux map as per Table 4.1-1 to update the R for each selected thimble.

(3) Base Load Operation

1. Base Load operation may be used at power levels between P_T and P_{BL} or P_T and 1.00 (whichever is most limiting). The maximum relative power permitted under Base Load operation,

P_{BL}, is equal to the minimum value of the ratio of $[F_Q(Z)]_L/[F_Q(Z)]_{BL}^{Meas}$ where $[F_Q(Z)]_{BL}^{Meas}$ is equal to

$[F_Q(Z)]_{Map}^{Meas}$ x W(Z) x 1.09, and $[F_Q(Z)]_L$ is equal to $[F_Q]_L$ X K(Z).

For the purpose of the specification, $[F_Q(Z)]_{Map}^{Meas}$ shall

be obtained between the elevations bounded by $\pm$ 10% of the active core height. The function W(Z) is determined analytically and accounts for the most perturbed power shapes which can occur under the constraints of Section 3.2.6.a(3)4. W(Z) corresponding to either $\pm$ 2% or $\pm$ 3% ΔI may be used to infer P_{BL}. The uncertainty factor of 9.0% accounts for manufacturing tolerances, measurement error, rod bow, and any burnup and power dependent peaking factor increases. Base Load operation can be utilized only if Section 3.2.6.a(3)2 or Section 3.2.6.a(3)3 is satisfied.

2. NOTE: For entering Base Load operation with power less than P_T.

   Prior to going to Base Load operation, maintain the following conditions for at least 24 hours:

   (1) Relative power must be maintained between $P_T/1.05$ and P_T.

   (2) ΔI within $\pm$2% or $\pm$ 3% ΔI target band for at least 23 hours per 24 hour period. The corresponding W(Z) is to have been used to determine P_{BL}.

   After 24 hours have elapsed a full core flux map to determine $[F_Q(Z)]_{Map}^{Meas}$ shall be taken unless a valid full core flux map was taken within the time period specified in Section 4.1. P_{BL} is then to be calculated as per Section 3.2.6.a(3)1.

3. NOTE: For entering Base Load operation with power greater than P_T.

   Prior to going to Base Load operation and prior to discontinuing augmented surveillance of hot channel factors, maintain the following conditions for at least 24 hours:

   (1) Relative power must be maintained between P_T and the power limited by augmented surveillance of hot channel factors.

   (2) ΔI within $\pm$ 2% or $\pm$ 3% ΔI target band. Corresponding W(Z) to have been used to determine P_{BL}.

   After 24 hours have elapsed a full core flux map to determine $[F_Q(Z)]_{Map}^{Meas}$ shall be taken unless a valid full core flux map was taken within the time period specified in Section 4.1. P_{BL} is then to be calculated as per Section 3.2.6.a(3)1.

4. If the conditions of Section 3.2.6.a(3)2 <u>or</u> of Section 3.2.6.a(3)3 are satisfied, then Base Load operation <u>may</u> be utilized provided the following is maintained.

   (1) Power between P_T and P_{BL} or P_T and 1.00 (whichever is most limiting).

   (2) ΔI within $\pm$ 2% or $\pm$ 3% ΔI target band. Corresponding W(Z) to have been used to determine P_{BL}.

   (3) Subsequent full core flux maps are taken within the time period specified in Section 4.1.

5. If any of the requirements of Section 3.2.6.a(3)4 are not maintained, then power shall be reduced to less than or equal to P_T, or within 15 minutes augmented surveillance of hot channel factors shall be initiated if the power is above P_T.

(4) Radial Burndown Operation

1. Radial Burndown operation is restricted to use at powers between P_T and P_{RB} or P_T and 1.00 (whichever is most limiting). The maximum relative power permitted under Radial Burndown operation, P_{RB}, is equal to the minimum value of the ratio of $[F_Q(Z)]_L/[F_Q(Z)]_{RB}^{Meas}$ where $[F_Q(Z)]_{RB}^{Meas} = [F_{xy}(Z)]_{Map}^{Meas} \times F_z(Z) \times 1.09$, and $[F_Q(Z)]_L$ is equal to $[F_Q]_L] \times K(Z)$

2. A full core flux map to determine $[F_{xy}(Z)]_{Map}^{Meas}$ shall be taken within the time period specified in Section 4.1.

   For the purpose of the specification, $[F_{xy}(Z)]_{Map}^{Meas}$ shall be obtained between the elevations bounded by $\pm$ 10% of the active core height.

3. The function $F_z(Z)$ is determined analytically and accounts for the most perturbed axial power shapes which can occur under axial power distribution control. The uncertainty factor of 9% accounts for manufacturing tolerances, measurement error, rod bow, and any burnup dependent peaking factor increases.

4. Radial Burndown operation may be utilized at powers between P_T and P_{RB} or P_T and 1.00 (whichever is most limiting) provided that the indicated flux difference is within $\pm$ 5% of the target axial offset.

5. If any of the requirements of Section 3.2.6.a(4)4 are not maintained, then the power shall be reduced to less than or equal to P_T or within 15 minutes augmented surveillance of hot channel factors shall be initiated if the power is above P_T.

b. (1) The measurement of total peaking factor, $[F_Q(Z)]^{Meas}_{Map}$ shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error. These uncertainties only apply if the map is taken for purposes other than determination of P_{BL} and P_{RB}.

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

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

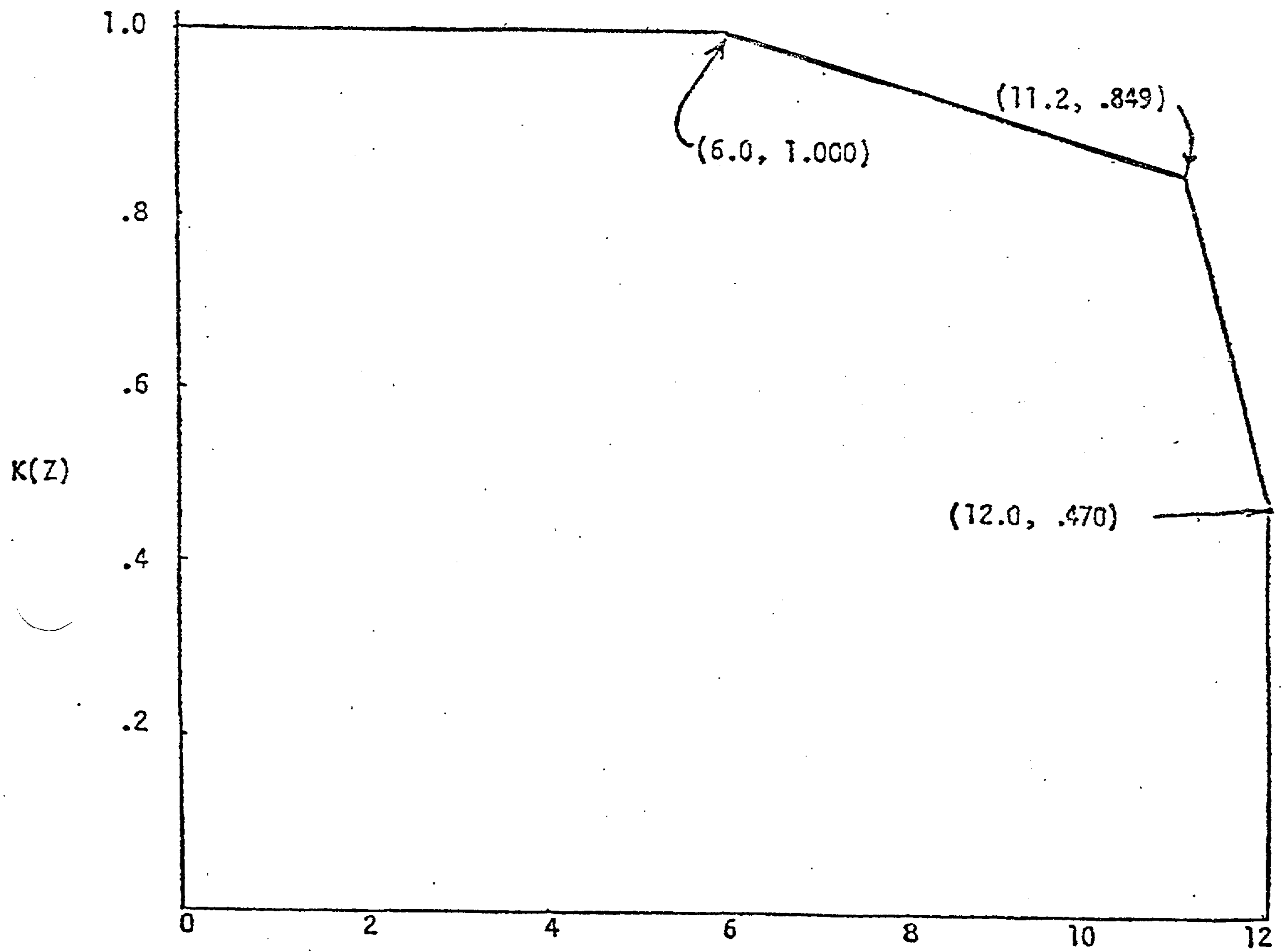
c. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter. If the axial flux difference has not been measured in the last effective full power month, the target flux difference 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). During Baseload Operation (Section 3.2.6.a(3)), the indicated axial flux shall be maintained within a ± 2% or ± 3% band about the target flux difference.

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

HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

(for $\leq$ 28% steam generator tube plugging and $[F_Q]_L$ = 2.125)



CORE HEIGHT (FT.)

TABLE 4.1.-1
MINIMUN FREQUENCIES FOR CHECKS, CALIBRATIONS AND
TEST OF INSTRUMENT CHANNELS

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
1. a.	Nuclear Power Range (Check, Calibrate and Test only applicable above 10% of rated power.)	S(1) M*(4)	D(2) Q*(4)	M(3)	1) Load vs. flux curve, or ΔT vs. reactor power curve 2) Thermal power calculation 3) Signal to ΔT, bistable action (permissive, rod stop, trips) 4) Upper & lower detectors for symmetric offset (+5 to -5%).
b.	Power Distribution Map			M(1) (2) (3)	1) Following initial loading and prior to operation above 75% power. 2) Once per effective full power month. 3) Confirm hot channel factors within limits.
2.	Nuclear Intermediate Range	$S(1)^+$	N. A.	P(2)	1) Once/shift up to 50% R.P. 2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S(1)	N. A.	P(2)	1) Once/shift when in service. 2) Disable action (alarm, trip)
4.	Reactor Coolant Temperature	S^+	R	$B/W(1)^+$ $(2)^+$	1) Overtemperature Δ T 2) Overpower ΔT
5.	Reactor Coolant Flow	S^+	R	M^+	
6.	Pressurizer Water Level	S^+	R	M^+	
7.	Pressurizer Pressure	S^+	R	M^+	
8.	4 kv Voltage & Frequency	N. A.	R**	R	Reactor protection circuits only
9.	Analog Rod Position	S^+	R	M^+	With step counters.

6.9.3 SPECIAL REPORTS

Special reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate.

Twenty copies of the following reports should be sent to the Director, Nuclear Reactor Regulation.

a. In-service inspection, reference 4.2.

b. Tendon surveillance, reference 4.4.

c. Fire protection systems, reference 3.14.

d. Peaking Factor Limit Report - The W(Z) function(s) for Base-Load Operation corresponding to a ±2% band about the target flux difference and/or a ±3% band about the target flux difference, the Load-Follow function $F_Z(Z)$ and the augmented surveillance turnon power fraction, P_T, shall be provided to the Director, Nuclear Reactor Regulations, Attention Chief of the Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 at least 60 days prior to cycle initial criticality, whenever P_T is <1.0. In the event, the option of Baseload Operation (as defined in Section 3.2.6.a [3]) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_Z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the values would be submitted 60 days prior to the date the values would become effective unless otherwise approved by the Commission.

6.9.4 UNIQUE REPORTING REQUIREMENTS

a. Radioactive Effluent Releases

A report of the quantities of radioactive effluents released from the plant, with data summarized on a monthly basis following the format of U.S. NRC Regulatory Guide 21.

The report shall be submitted within 60 days after January 1 and after July 1 specifying quantities of radioactive effluents released during the previous 6 months of operation.

1. Gaseous Releases

(a) Total radioactivity (in curies) releases of noble and activation gases.

(b) Maximum noble gas release rate during any one-hour period.

(c) Total radioactivity (in curies) released by nuclide, based on representative isotopic analyses performed.

(d) Percent of technical specification limit.

2. Iodine Releases

(a) Total (I-133, I-135) radioactivity (in curies) released.

(b) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

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.

$F_{\Delta H}^N$, 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 as defined by 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.

The results of the loss of coolant accident analyses based on this upper bound envelope indicate a peak clad temperature could theoretically exceed the 2200°F limits. To ensure the criteria are not violated, MIDS will be used to provide a more exact indication of F_Q. Note that MIDS and a penalty on F_Q are only required above P_T to meet the acceptance criteria as justified in the analyses. Below P_T, the nuclear analyses of credible power shapes consistent with these specifications have shown that the limit of $[F_Q]_L/P$ times Figure 3.2-3 is not exceeded provided the limits of Figure 3.2-3 are applied.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate 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. These uncertainties only apply if the map is taken for purposes other than the determination of P_{BL} and P_{RB}.

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) although 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 prediction 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 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.

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 Differences ($\Delta\phi$) and a reference value which corresponds to the full 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 assures that the $[F_Q]_L$ upper bound envelope as defined by 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 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 further as burnup proceeds). This value, divided by the fraction of full power at which the core was operating, is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviations 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 differences 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-

*Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

tions 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 observed, is not worse than a 10 percent increment in peaking factor for flux differences 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 rated 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 rated power is required to protect against potentially more severe consequences of some accidents.

The analytically determined $[F_Q]_D$ is formulated to generate limiting shapes for all load follow maneuvers consistent with control to a $\pm$ 5% band about the target flux difference. For Base Load operation the severity of the shapes that need to be considered is significantly reduced relative to load follow operation. The severity of possible shapes is small due to the restrictions imposed by Sections 3.2.6.a(3)2, 3.2.6.a(3)3 and 3.2.6.a(3)4. To quantify the effect of the limiting transients which could occur during Base Load operation, the function W(Z) is calculated from the following relationship:

$$W(Z) = \text{Max}\left(\frac{F_Q(Z)\left(\text{Base Load Case(s), 150 MWD/T}\right)}{F_Q(Z)\left(\text{ARO, 150 MWD/T}\right)}, \frac{F_Q(Z)\left(\text{Base Load Case(s), 85\% EOL BU}\right)}{F_Q(Z)\left(\text{ARO, 85\% EOL BU}\right)} \right)$$

For Radial Burndown operation the full spectrum of possible shapes consistent with control to a $\pm$ 5% ΔI band needs to be considered in determining power capability. Accordingly, to quantify the effect of the limiting transients which could occur during Radial Burndown operation, the function $F_Z(Z)$ is calculated from the following relationship:

$$F_Z(Z) = [F_Q(Z)]_{\text{FAC Analysis}} / [F_{xy}(Z)]_{\text{ARO}}$$

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.

Above the power level of P_T, additional flux shape monitoring is required. In order to assure that the total power peaking factor, F_Q, is maintained at or below the limiting value, the movable incore instrumentation will be utilized. Thimbles are selected initially during startup physics tests so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor F_Q can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e. an effective radial peaking factor $\overline{R}$, can be determined as the ratio of the total peaking factor results from a full core flux map and the axial peaking factor in a selected thimble.

* Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

REFERENCES

FSAR - Section 14.3.2

The limiting value of $[F_j\ (Z)]_S$ is derived as follows:

$$[F_j\ (Z)]_S = \frac{[F_Q]_L\ [K(Z)]}{P_L \bar{R}_j\ (1+\sigma)\ (1.03)\ (1.07)}$$

Where:

a) $F_j\ (Z)$ is the normalized axial power distribution from thimble j at elevation Z.

b) P_L is reactor thermal power expressed as a fraction of 1.

c) K (Z) is the reduction in limit as a function of core elevation (Z) as determined from Figure 3.2-3.

d) $[F_j(Z)]_S$ is the alarm setpoint for MIDS.

e) $\bar{R}_j$, for thimble j, is determined from n=6 incore flux maps covering the full configuration of permissible rod patterns at the thermal power excore limit of P_T

$$\bar{R}_j = \frac{\sum_{i=1}^{n} R_{ij}}{n}$$

where

$$\bar{R}_{ij} = \frac{F_{Qi}^{meas}}{[F_{ij}(Z)]^{MAX}}$$

and $F_{ij}\ (Z)$ is normalized axial distribution at elevation Z from thimble j in map i which has a measured peaking factor without uncertainties or densification allowance of F_{Qi}^{Meas}.

f) σ_j is the standard deviation, expressed as a fraction or percentage of $\bar{R}j$, and is derived from n flux maps and the relationship below, or 0.02 (2%), whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1}\sum_{i=1}^{n}(R_{ij}-\bar{R}_j)^2\right]^{1/2}}{\bar{R}_j}$$

g) The factor 1.03 reduction in the Kw/ft limit is the engineering uncertainty factor.

h) The factors $(1 + \sigma_j)$ and 1.07 represent the margin between $[F_j\ (Z)]_L$ limit and the MIDS alarm setpoint $[F_j(Z)]_S$. Since $(1 + \sigma_j)$ is bounded by a lower limit of 1.02, there is at least a 9% reduction of the alarm setpoint. Operations are permitted in excess of the operational limit $\leq$ 4% while making power adjustment on a percent for percent basis.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NO. DPR-31
AND AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-41
FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT PLANT UNIT NOS. 3 AND 4
DOCKET NOS. 50-250 AND 50-251

Introduction

By letter dated May 14, 1981, as supplemented on November 23, 1981 and January 28, 1982, the Florida Power and Light Company (the licensee) requested amendments to Facility Operating License Nos. DPR-31 and DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4. The amendments would change the Technical Specifications to specify new power distribution limits for base load and radial burndown operation.

We have found it necessary to modify the amendment as proposed. We discussed the modifications with the licensee staff. They have agreed to the modifications and the modifications have been incorporated.

Discussion

The Technical Specifications for Westinghouse designed power plants contain limits on the total heat flux peaking factor, $F_Q \times K(z)$. These limits are usually established by the LOCA analysis. Generally, assurance that the limits are not exceeded in normal operation of the power plant is demonstrated by either a generic[1] or plant specific analysis[2] of permissible load following maneuvers. The generic analysis applies when the limit is $2.32 \times K(z)$, and the plant specific analysis when the F_Q portion of the limit is less than 2.32. If all the points predicted by the analysis fall below the $F_Q \times K(z)$ limit, then the limit will not be exceeded, if operation of the power plant is in conformance with the assumptions used in the analysis. The power distribution Technical Specifications have been written to ensure this conformance.

The subject change request retains the option discussed in the preceding paragraph and also adds Specifications for three additional modes to ensure that the $F_Q \times K(z)$ limits will not be exceeded when the analytically predicted peaking factor values exceed the $F_Q \times K(z)$ limits. The three modes are called MIDS, Base Load Operation and Radial Burndown Operation.

Evaluation

In the MIDS mode, incore detectors are used to measure the actual peaking factors in the reactor, at a power level fraction (or %) above the limiting ratio of the $F_Q \times K(z)$ curve to the analytically predicted peaking factors. This power level is designated as P_T. The MIDS mode is a manual application of the Westinghouse axial power distribution monitoring system (APDMS). The APDMS is an approved technology[3]. A form of MIDS has been approved for and n use at the Turkey Point reactors and other reactors for a number of years. The proposed MIDS Technical Specifications are therefore acceptable.

It should be pointed out that there are a few slight differences between the MIDS requirements and a generic APDMS Specification. In MIDS, incore traces are initiated (in addition to when the power level is raised above P_T) when control bank D is moved more than an accumulated total of 15 steps in one direction. The generic action occurs at 5 steps. The licensee has provided data in his submittal showing that the change in the axial flux profile would be almost negligible for a rod motion of 5 steps at typical rod insertions used in the Turkey Point reactors. Use of the larger motion to initiate traces is therefore acceptable.

Because the Turkey Point reactors are base loaded, the licensee also proposed to reduce the interval for incore traces from the 8 hour generic value to 24 hours. We find the frequency is acceptable for reactors which are primarily base loaded. If operating experience indicates the $F_Q \times K(z)$ limits are exceeded with this frequency of surveillance, we will take action to require more frequent traces. Furthermore, the Specification provides for more frequent traces under any load swing which reduces power below and then above P_T again. This is the most important time for frequent surveillance.

The licensee has also proposed a sequence of traces at least 2, 4, 8 and every 24 hours thereafter following the need to initiate traces. This eliminates an immediate trace, and traces at ½ and 1 hour sometimes found in APDMS Specifications. We find the proposed trace surveillance schedule acceptable for the Turkey Point Units, because the units essentially do not load follow, so that the peaking factor variations will be small.

In the Base Load Operation mode, the predicted peaking factors described in e first (normal) mode under Discussion above, are replaced with an axially uependent set of peaking factors which are the product of measured steady state peaking factors (measured every effective full power month), pre-calculated transient factors (W(z)) and appropriate uncertainties. The most limiting ratio of the $F_Q \times K(z)$ limit to this product defines a power fraction P_{BL}. Base Load Operation can then be used between P_T and P_{BL} or 1.00, whichever is most limiting. This technique has also been approved[4] and is in use at three reactor sites. We have reviewed the Westinghouse derived set of cases used to generate the W(z) function, and as a result of our experience with load following analyses find them acceptable, because they conservatively bound conditions which will be encountered in normal operation of the power plant.

The uncertainty used for the Base Load Operation mode is 1.09. It is obtained from a statistical combination of the nuclear uncertainty, engineering uncertainty, a conservative allowance for peaking factor uncertainty from rod bowing, and an uncertainty to account for a possible nup or power dependent peaking factor increase between the effective full power monthly maps to determine the steady state peaking factor. The licensee has provided information on the independence of these uncertainties a necessary condition for their statistical combination. This is the scientifically appropriate way to treat uncertainties, and we therefore find it acceptable for use in this application. Our review of the details of Base Load Operation provided for Turkey Point Units 3 and 4, including restriction of operation to a narrow delta flux band, is favorable as discussed above. Because of this, and since modes similar to Base Load Operation have been approved and are in use at other operating reactor sites, we find the proposed Base Load Operation Technical Specifications acceptable.

The last mode, Radial Burndown Operation, uses the predicted limiting axial peaking factors from the first (normal) mode, but combines then (as a product) with the uncertainty used in Base Load Operation and with planar peaking factors measured every effective full-power month rather than the predicted planar peaking factors used in the first mode. If the actual planar peaking factors are smaller than predicted, or burndown during the cycle (both of these trends usually occur), then Radial Burndown Operation can provide an advantage in allowable power level above the first mode, even though the uncertainty allowance is slightly larger because of the inclusion of the factor to account for burnup or power dependent peaking factor increases. The Radial Burndown Operation mode is applicable between power levels of P_T and P_{RB}, which is the ratio of the F_Q x K(z) limit to the peaking factor calculated for this mode, or 1.00, whichever is most limiting. We have approved variations of operating modes similar to Radial Burndown Operation for other reactors. In fact, this option merely combines elements from the first mode (the predicted limiting axial peaking factors) and the uncertainty and measured planar peaking factors (which is part of the measured steady state peaking factor) from the Base Load Operation mode. Radial Burndown Operation is therefore acceptable because it combines elements of otherwise approved techniques.

The proposed Technical Specifications for the three modes adequately provide for surveillance of the peaking factors and return to a conservative state, or in case none of the options is usable, a return to the power level P_T, if the F_Q x K(z) limit is exceeded.

The proposed Technical Specification changes also require submittal of a Peaking Factor Limit Report in Section 6.9.3. This report must be submitted 60 days before it is needed to the NRC. The report will contain the W(z) functions for Base Load Operation, the predicted limiting axial peaking factors for Radial Burndown Operation and the augmented surveillance turnon power fraction, P_T. We find provision of these quantities in this manner acceptable because it will eliminate routine cycle dependent changes to the Technical Specifications, but will provide us with the specified information. We would then be able to obtain further information if any trend in the data became a matter of concern.

We have performed a careful review of the interrelationships of the proposed Technical Specifications, and find them all acceptable. For the reasons stated above, each of the peaking factor surveillance modes is acceptable. We conclude that surveillance in these modes will continue to provide assurance that the peaking factors used as initial assumptions for the LOCA analysis will not be violated in normal operation of the power plant, and thus the proposed changes to the Technical Specifications will not substantially reduce the safety margins maintained in the power plants, nor adversely affect the health and safety of the public.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the consideration discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: March 17, 1982

Principal Contributor:
M. Dunenfeld

REFERENCES

1. T. Morita, et al., "Power Distribution Control and Load Following Procedures," WCAP-8385, Westinghouse Electric Corporation, September 1974.

2. Letter from Westinghouse (C. Eicheldinger) to NRC (J. F. Stolz) dated April 6, 1978, Serial No. 986C.

3. K. A. Jones, et al., "Axial Power Distribution Monitoring System," WCAP-8589, Westinghouse Electric Corporation, August 1975.

4. J. Holm, R. J. Burnside, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors - Phase 2" XN-NF-77-57(A) Exxon Nuclear Corporation, May 1981.

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

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 80 to Facility Operating License No. DPR-31, and Amendment No. 74 to Facility Operating License No. DPR-41 issued to Florida Power and Light Company (the licensee), which revised Technical Specifications for operation of Turkey Point PLant, Unit Nos. 3 and 4 (the facilities) located in Dade County, Florida. The amendments are effective as of the date of issuance.

The amendments change the Technical Specifications to specify new power distribution limits related to base load and radial burndown operation.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated May 14, 1981, as supplemented on November 23, 1981 and January 28, 1982, (2) Amendment Nos. 80 and 74 to License Nos. DPR-31 and DPR-41, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Environmental and Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 17th day of March, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

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