3/3/78

Docket Nos. ~~50-250~~
and 50-251

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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 Nos. 33 and 27 to Facility Operating License Nos. DPR-31 and DPR-41 for the Turkey Point Nuclear Generating Units Nos. 3 and 4. The amendments consist of changes to the Technical Specifications in response to your application dated January 27, 1978, supplemented by letters dated February 15 and February 17, 1978.

These amendments authorize operation of Turkey Point Units Nos. 3 and 4 with up to an average of 19% of the tubes in the three steam generators in a plugged condition.

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

Sincerely,

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 33 to DPR-31
2. Amendment No. 27 to DPR-41
3. Safety Evaluation
4. Notice

cc w/enclosures:
see next page

OT: RSB
R. Baer
3/3/78

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DATE	3/01/78	3/01/78	3/2/78	3/3/78	3/2/78	3/3/78

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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 NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
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 January 27, 1978, as supplemented by letters dated February 15 and February 17, 1978, 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. 33, 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

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 33

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-31

DOCKET NO. 50-250

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised page is identified by Amendment number and contains vertical lines indicating the area of change.

Remove	Replace
Figure 2.1-1	Figure 2.1-1
-	Figure 2.1-1a (new)
2.3-2	2.3-2
2.3-3	2.3-3
3.1-7	3.1-7
3.2-3	3.2-3

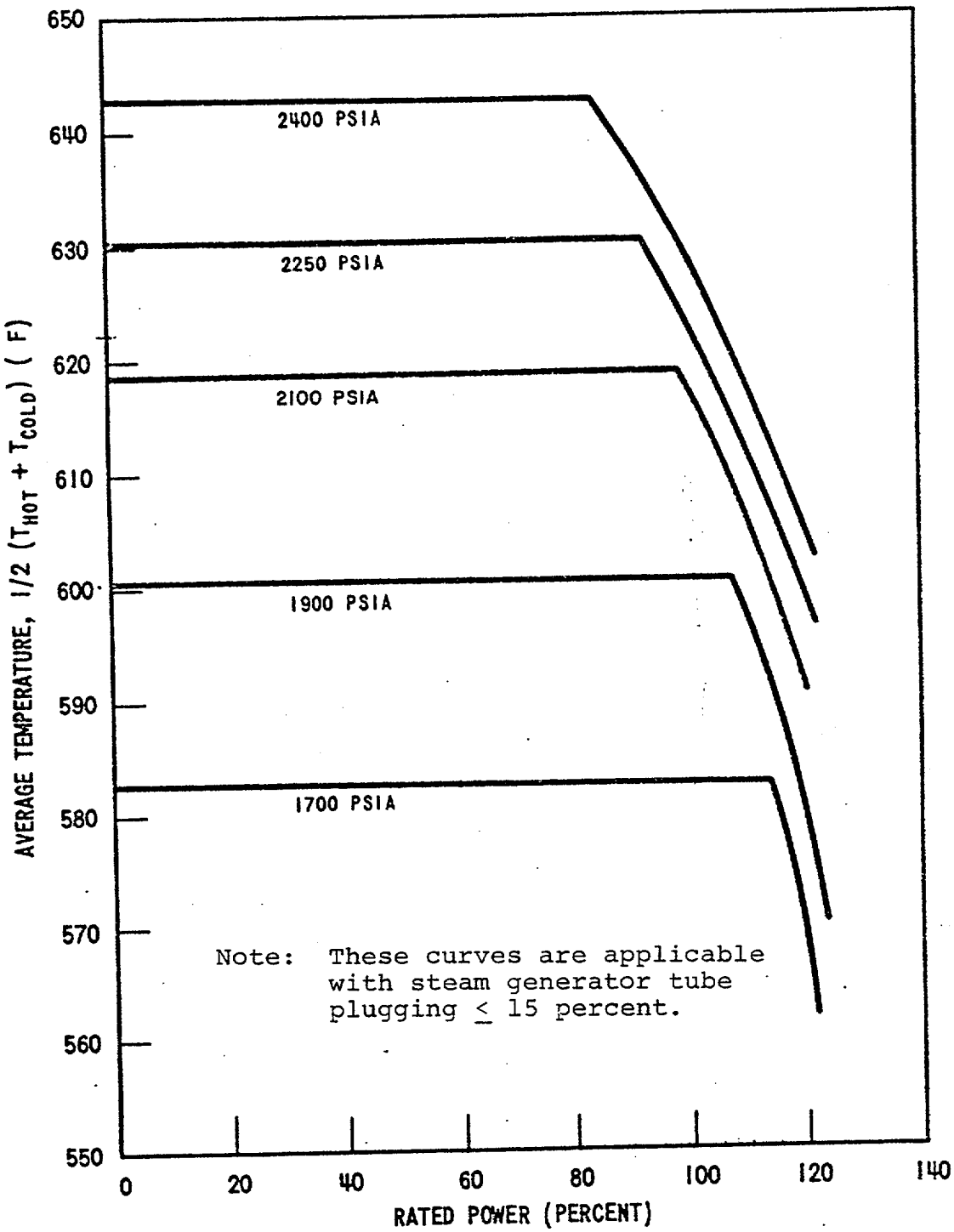


Figure 2.1-1. Reactor Core Thermal and Hydraulic Safety Limits, Three Loop Operation.

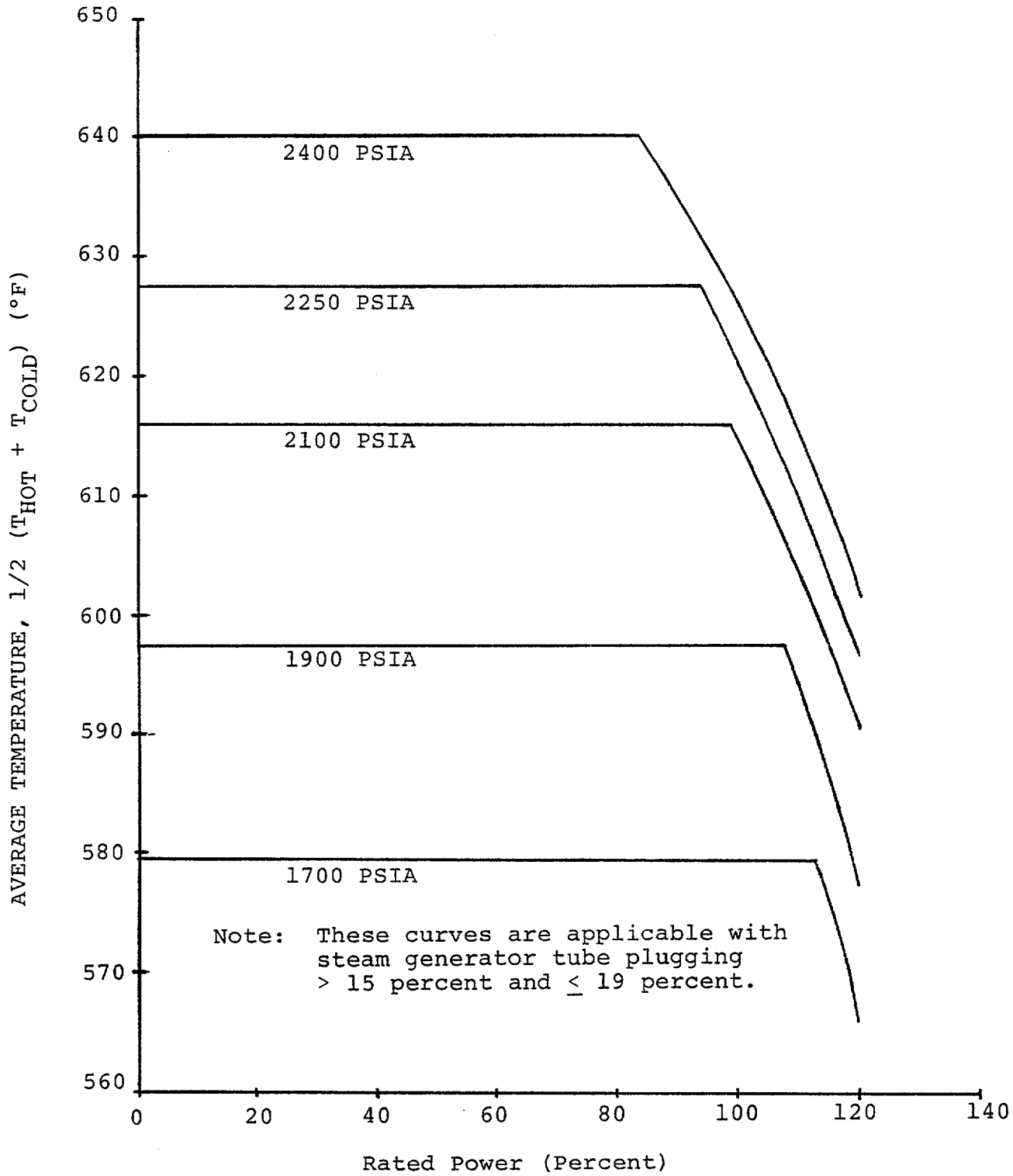


Figure 2.1-1a Reactor Core Thermal and Hydraulic Safety Limits, Three Loop Operation

Reactor Coolant Temperature

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

ΔT_o = Indicated ΔT at rated power, 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) = 1.095*

(Two Loop Operation) = 0.88

---

*K_1 = 1.095 for steam generator tube plugging $\leq$ 15 percent

K_1 = 1.08 for steam generator tube plugging > 15 percent and $\leq$ 19 percent

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

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

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

---

*This factor is 1.11 for steam generator tube plugging $\leq$ 15 percent.

This factor is 1.10 for steam generator tube plugging > 15 percent and $\leq$ 19 percent.

6. DNB PARAMETERS

The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg $\leq$ 578.2°F

b. Pressurizer Pressure $\geq$ 2220 psia*

c. Reactor Coolant Flow $\geq$ 268,500 gpm†

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limits after each refueling cycle.

---

* Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

† Reactor Coolant Flow $\geq$ 268,500 gpm for steam generator tube plugging $\leq$ 15 %.

Reactor Coolant Flow $\geq$ 263,130 gpm for steam generator tube plugging > 15% and $\leq$ 19%.

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. Hot channel factors:

1. With steam generator tube plugging $\leq$ 15%, the hot channel factors (defined in the basis) must meet the following limits at all times except during low power physics tests:

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

$$F_q\ (Z) \leq (4.44) \times K(Z), \text{ for } P \leq .5$$

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

Where P is the fraction of rated power at which the core is operating; K(Z) is the function given in Figure 3.2-3; Z is the core height location of F_q.

2. With steam generator tube plugging > 15% and $\leq$ 19%, the hot channel factors must meet the following limits at all times except during low power physics tests:

$$F_q\ (Z) \leq (2.05/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \leq (4.10) \times K(Z), \text{ for } P \leq .5$$

$$F^N_H \leq 1.55\ [1 + 0.2\ (1-P)]$$

Where P, K(Z), and Z are defined in 1. above.

If predicted F_q exceeds 2.05 with tube plugging > 15% and $\leq$ 19%, then power will be limited to the rated power multiplied by the ratio of 2.05 divided by the predicted F_q, or augmented surveillance of hot channel factors shall be implemented.

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,

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
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 January 27, 1978, as supplemented by letters dated February 15 and February 17, 1978, 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 by 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. 27, 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

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 27

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-251

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised page is identified by Amendment number and contains vertical lines indicating the area of change.

Remove	Replace
Figure 2.1-1	Figure 2.1-1
-	Figure 2.1-1a (new)
2.3-2a	2.3-2
2.3-3a	2.3-3
3.1-7a	3.1-7
3.2-3	3.2-3

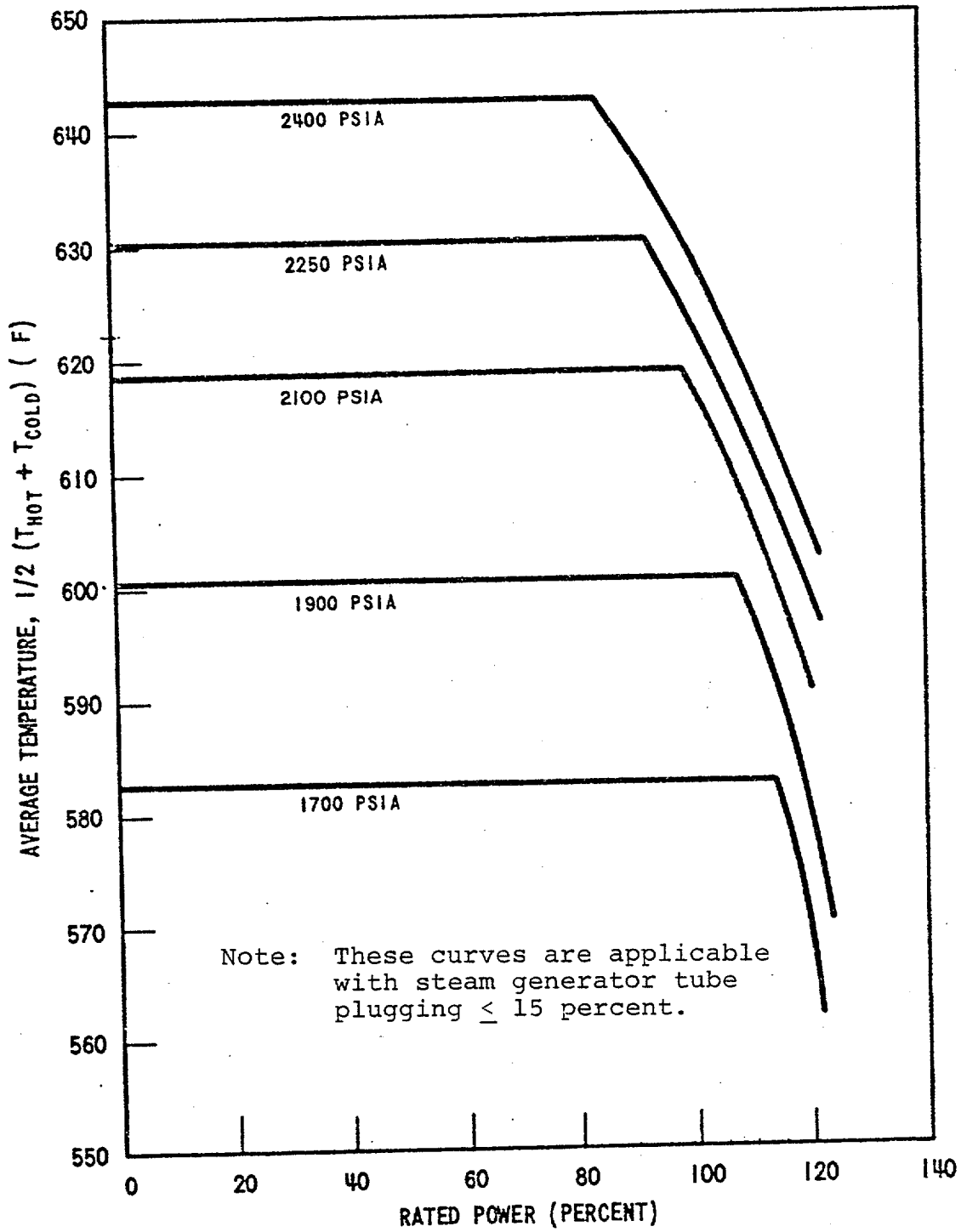
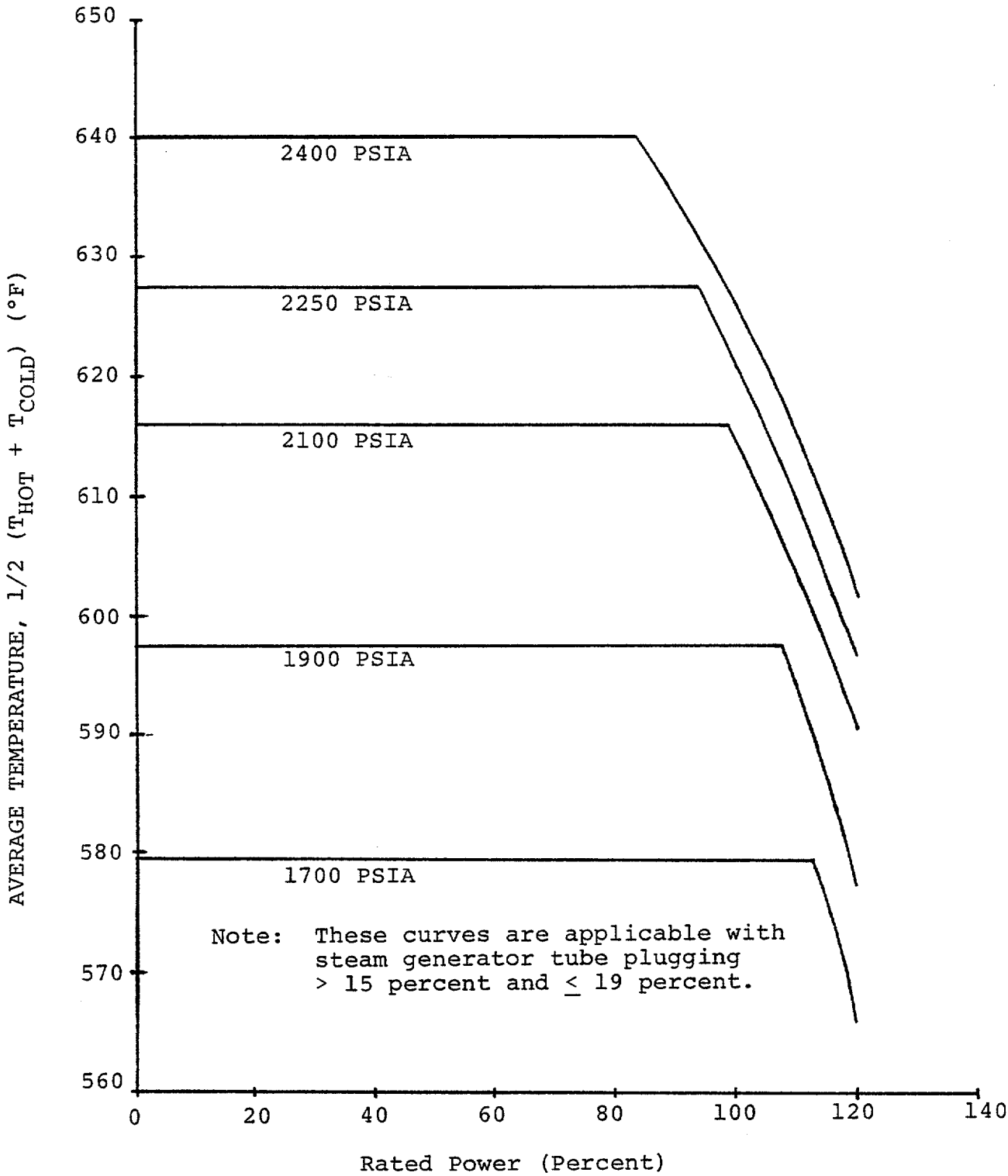


Figure 2.1-1. Reactor Core Thermal and Hydraulic Safety Limits, Three Loop Operation



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ΔT_o = Indicated ΔT at rated power, F

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K_1 (Three Loop Operation) = 1.095*

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---

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This factor is 1.10 for steam generator tube plugging > 15 percent and $\leq$ 19 percent.

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The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg $\leq$ 578.2°F

b. Pressurizer Pressure $\geq$ 2220 psia*

c. Reactor Coolant Flow $\geq$ 268,500 gpm†

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limits after each refueling cycle.

---

* Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

† Reactor Coolant Flow $\geq$ 268,500 gpm for steam generator tube plugging $\leq$ 15 %.

Reactor Coolant Flow $\geq$ 263,130 gpm for steam generator tube plugging > 15% and $\leq$ 19%.

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
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a. Hot channel factors:

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$$F_q\ (Z) \leq (4.44) \times K(Z), \text{ for } P \leq .5$$

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

Where P is the fraction of rated power at which the core is operating; K(Z) is the function given in Figure 3.2-3; Z is the core height location of F_q.

2. With steam generator tube plugging > 15% and $\leq$ 19%, the hot channel factors must meet the following limits at all times except during low power physics tests:

$$F_q\ (Z) \leq (2.05/P) \times K(Z), \text{ for } P > .5$$

$$F_q\ (Z) \leq (4.10) \times K(Z), \text{ for } P \leq .5$$

$$F^N_H \leq 1.55\ [1 + 0.2\ (1-P)]$$

Where P, K(Z), and Z are defined in 1. above.

If predicted F_q exceeds 2.05 with tube plugging > 15% and $\leq$ 19%, then power will be limited to the rated power multiplied by the ratio of 2.05 divided by the predicted F_q, or augmented surveillance of hot channel factors shall be implemented.

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,

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 33 AND 27 TO LICENSE NOS. DPR-31 AND DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNITS NOS. 3 AND 4

DOCKETS NOS. 50-250 AND 50-251

Introduction

By application dated January 27, 1978 (supplemented by letters dated February 15, 1978 and February 17, 1978) the Florida Power and Light Company (FPL) requested amendments to Operating Licenses Nos. DPR-31 and DPR-41 for the Turkey Point Plant Units Nos. 3 and 4. The application, which contains accident analyses and Technical Specification changes is in support of a request to raise the steam generator tube plugging level limit from 15% to 19% for each unit.

Reactor Coolant System Flow Rate

As the level of steam generator tube plugging increases several factors affect the assumptions used for the analyses of anticipated and design basis accidents. Among the affected parameters are the pump coast down rate, heat transfer area to the secondary side, and loop flow rate. To assess the effect of steam generator tube plugging on reactor coolant system (RCS) loop flow, FPL has taken measurements to obtain the loop flow rate at the 13.2% level of steam generator tube plugging. The flow rate at this level of steam generator tube plugging was measured to be 103.4% of the thermal design flow.

By use of the relationship of 1% flow reduction per 3% tube plugging FPL has extrapolated to the flow rate at 19% plugging. The predicted flow rate at this plugging level is 101.5% of the thermal design flow. To add conservatism this flow rate is then reduced by 3.5% to 98% of the thermal design flow. Error analyses performed on flow measurements made at other operating PWR's, such as Surry, Oconee, and Three Mile Island, have yielded errors on the order of 2%. Therefore, a reduction of 3.5% in the estimated flow rate is conservative.

Transient and Accident Analyses

A. Non-LOCA Accidents

As a result of the higher plugging level three factors made it necessary to reexamine the transients reported in the FSAR:

1. The RCS flow rate will be lower due to the reduced flow area,

2. The RCS volume will be less due to the reduced steam generator tube volume,

3. The pump coast down characteristics will be more severe.

FPL has submitted(3) an assessment of the impact on the non-LOCA events of steam generator tube plugging up to a level of 19%. The individual events were examined as to which of the parameters affected by increased steam generator tube plugging were important for each accident. These affected events which are limiting or very sensitive to the effects of higher steam generator tube plugging levels were evaluated. The following assumptions were used in the analyses:

	Proposed	Current
Thermal design flow, gpm/loop	89,500 to 81,400	89,500
S. G. tube plugging, %	19	15
Power, Mwt (102% of)	2200	2200
T_{IN}, °F (+ 4°F)	550.2	550.2
ΔT at 100% Power, °F	57.1	57.1
$F^N_{\Delta H}$	1.55	1.55
F_Q	2.32	2.32

Two accidents are limiting or most sensitive to the higher steam generator tube plugging level:

1. An uncontrolled control rod assembly withdrawal at power produces a mismatch in reactor power and steam flow. The result is an increase in reactor coolant temperature. Reevaluation of the uncontrolled withdrawal at power indicates that adequate margin to DNB would exist at the higher steam generator tube plugging level. The minimum DNBR calculated remains above 1.30, thus indicating that the core and reactor coolant system would not be adversely affected.

2. The loss of flow accident was reevaluated on the basis of loss of all three reactor coolant pumps. The significant factor affecting the loss of flow accident is the increased loop resistances, due to the higher level of steam generator tube plugging, resulting in a more rapid pump coastdown.

   Reevaluation of the loss of flow event for the higher level of steam generator plugging results in a minimum DNBR greater than 1.30. Thus, adequate margin exists for the loss of flow event with a higher level of steam generator tube plugging.

The result of the evaluations and reanalyses performed indicate that with the reduced flow rate the anticipated transients presented in the FSAR will meet NRC requirements for safety margin.

B. LOCA

The ECCS performance has been reanalyzed(1) for a postulated large break with the assumed flow rate in Section II above. The reanalysis was performed using the October, 1975 version of the Westinghouse Evaluation Model. That model is in compliance with Appendix K to 10 CFR 50 and the August 27, 1976 Order for Modification of License.

The assumptions and initial operating conditions used in the reanalysis are in accord with those of the current approved LOCA-ECCS analysis with the exception of:

1. The total peaking factor changed from 2.32 to 2.05

2. RCS flow rate changed from 89,500 gpm/loop to 87,710 gpm/loop,

3. Number of steam generator tubes plugged changed from 15% to 19%,

4. Core inlet temperature uncertainty of +4°F removed,

Results have been submitted for the double ended cold leg guillotine break (DECLG) with a discharge coefficient C_D=0.4. As with previous analyses for the Turkey Point Units the break with C_D=0.4 is the limiting case. The results of the reanalysis indicate a peak clad temperature of 2195°F, a maximum local clad oxidation rate of 12.4%, and a total core metal-water reaction of less than 0.3%.

The results given above were obtained using a core power shape axially peaked at the core centerline. The Westinghouse ECCS sensitivity studies (4) indicate that the center peaked shape is the limiting power shape for peaking factors (F_Q) greater than 2.32. The staff has considered the applicability of the Westinghouse sensitivity study's result that the center peaked shape is limiting when the peaking factor is less than 2.32.

Amendment 66[5,6] to the D. C. Cook FSAR supports a peaking factor of 2.05. Although D. C. Cook is a different plant, the results of calculations performed with two power shapes, center and top peaked, demonstrate that at a peaking factor of 2.05 the center peaked shape remains limiting and tends to confirm the conclusion of the Westinghouse analyses. Similar considerations of the effect of peaking factor on the limiting power shape were also carried out on Surry, another three-loop plant. The results of these considerations[7] indicate that for a peaking factor as low as 1.85 the center peaked shape is still limiting. Therefore, we conclude that for Turkey Point Units Nos. 3 and 4 with a peaking factor of 2.05 the limiting case has been analyzed.

We conclude that the ECCS meets the Acceptance Criteria as presented in 10 CFR 50.46 with up to 19% of the steam generator tubes plugged.

Technical Specifications

The proposed Technical Specification changes have been reviewed to insure that the assumptions and limitations imposed due to the accident analyses and LOCA reanalyses will be met. Non-LOCA events have been analyzed with F_Q equal to 2.32. Reanalysis of the non-LOCA event with the LOCA limit of 2.05 would yield a higher DNBR value and increased safety margin. Therefore, the analysis was not repeated using an F_Q of 2.05. Although the non-LOCA events have been analyzed with the F_Q at 2.32, the Technical Specifications reflect the limiting F_Q value of 2.05 derived from the LOCA event. The proposed Technical Specifications will provide for safe operation of Turkey Point Units Nos. 3 and 4 with up to 19% of the steam generator tubes plugged. Therefore, we find the proposed Technical Specification changes acceptable.

Summary

The licensee has shown by the use of conservative assumptions and acceptable analyses that Turkey Point Units Nos. 3 and 4 can be safely operated with up to 19% of the steam generator tubes plugged. The limiting transients were reanalyzed and the results are within required safety margins. The ECCS performance has been reanalyzed for the large break LOCA with the result that compliance with the requirements of 10 CFR 50.46 is assured. For the limiting case the peak clad temperature does not exceed 2200°F.

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 this amendment.

Conclusion

We have concluded, based on the considerations 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.

Dated: March 3, 1978

References

1. Letter R. E. Uhrig (FPL) to V. Stello (NRC) dated January 27, 1978.

2. Letter R. E. Uhrig (FPL) to V. Stello (NRC) dated February 15, 1978.

3. Letter R. E. Uhrig (FPL) to V. Stello (NRC) dated February 17, 1978.

4. Salvatori, R., "Westinghouse ECCS-Plant Sensitivity Studies," WCAP-8356, July 1974, WCAP-8340 (Proprietary), July 1974.

5. D. C. Cook FSAR Amendment 66, November 1975.

6. Letter J. Tillinghast (IMPC) to B. Rusche (NRC) dated December 15, 1975.

7. Letter C. M. Stallings (Vepco) to E. G. Case (NRC) dated February 17, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-250 AND 50-251

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 33 and 27 to Facility Operating Licenses Nos. DPR-31 and DPR-41, respectively, issued to Florida Power and Light Company which revised Technical Specifications for operation of the Turkey Point Nuclear Generating Units Nos. 3 and 4, located in Dade County, Florida. The amendments are effective as of the date of issuance.

The amendment authorizes operation of Turkey Point Units Nos. 3 and 4 with up to an average of 19% of the steam generator tubes in each reactor plugged.

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 January 27, 1978, as supplemented by letters dated February 15 and February 17, 1978, (2) Amendments Nos. 33 and 27 to Licenses Nos. DPR-31 and DPR-41 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Environmental & Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 3 day of March 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors