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U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos. 50-250
and 50-251

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. 68 to Facility Operating License No. DPR-31 and Amendment No. 60 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 March 5, 1981.

These amendments incorporate the results of a revised ECCS analysis for a steam generator plugging level of 28%.

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

Sincerely,

Original signed by:
Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 68 to DPR-31
2. Amendment No. 60 to DPR-41
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD		
SURNAME	CParrish	MGrotenhuis	SVarga	TNovak	S. Goldberg		
DATE	4/16/81	4/16/81:ds	4/16/81	4/16/81	4/17/81		

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OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD		
NAME	CParrish	MGrotenhuis	SVarga	TNovak	[illegible]		
DATE	4/16/81	4/16/81:ds	4/16/81	4/16/81	4/17/81		

NRC FORM 318 (10-80) NRCM 0240

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Honorable Dewey Knight
County Manager of Metropolitan Dade County
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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 STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 68
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 March 5, 1981, 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.

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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. 68 , 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, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 23, 1981

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING STATION UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 60
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 March 5, 1981, 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. 60 , 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: June 23, 1981

REACTOR COOLANT TEMPERATURE

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

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

P = Pressurizer pressure, psig

f(Δq)= 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(Δq) = 0.

For each percent that the magnitude of (q_t - q_b) exceeds + 10 percent, the Delta-T trip setpoint 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 setpoint 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 $\leq$ 28 percent.

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

ΔT_o = Indicated T 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(Δq)=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 Presssurizer 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.

Undervoltage 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 15% of narrow range instrument scale.

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*This factor is 1.11 for steam generator tube plugging ≤15%.
This factor is 1.10 for steam generator tube plugging >15% and ≤19%.
This factor is 1.08 for steam generator tube plugging >19% and ≤28%.

+This factor is 0.00106 for steam generator tube plugging >19% and ≤28%.

6. <u>DNB PARAMETERS</u>

The following DNB related parameter 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 termal 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 pube plugging $\leq$15%.

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

Reactor Coolant Flow $\geq$ 255,075 gpm for steam generator tube plugging >19% and $\leq$28%.

reactivity insertion upon injection 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

(1) the rod cannot be moved by the CRDM, or
(2) the rod is misaligned from its bank by more than 15 inches, or
(3) 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 DISTRBUTION LIMITS

a. Hot channel factors:

With steam generator tube plugging $\leq$ 28%, the hot channel factors (defined in the basis) must meet the following limits at all times except during low power physics tests:

$$F_q (Z) \leq (2.125/P) \times K(Z), \text{ for } P > .5$$

$$F_q (Z) \leq (4.25) \times K(Z), \text{ for } P \leq .5$$

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

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.

If F_q, as predicted by approved physics calculations, exceeds 2.125 the power will be limited to the rated power multiplied by the ratio of 2.125 divided by the predicted F_q, or augmented surveillance of hot channel factors shall be implemented.

HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

(for $\leq$ 28% steam generator tube plugging and F_q = 2.125)

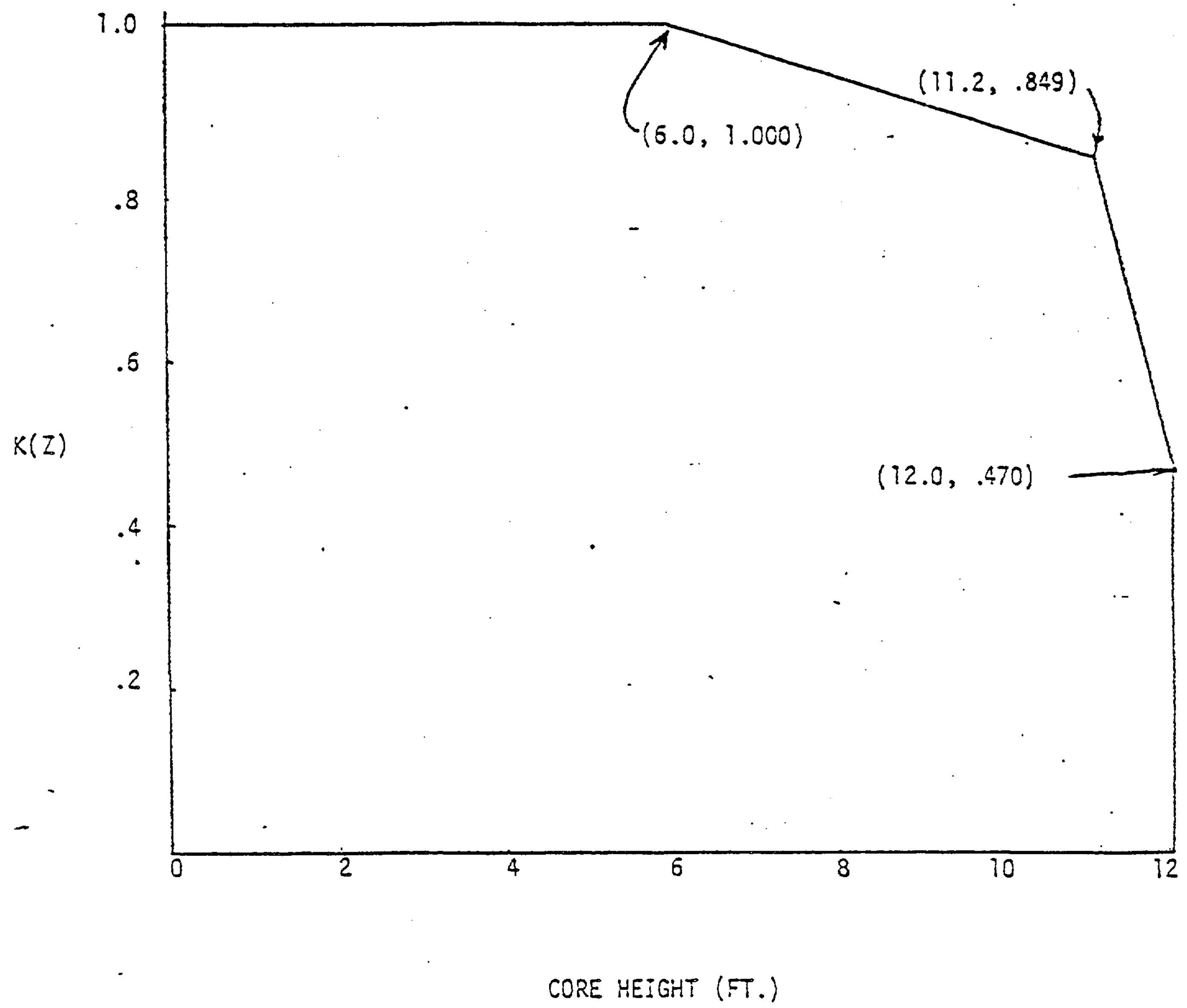

FIG. 3.2-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO. 60 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

I. Introduction

By letter dated March 5, 1981 Florida Power and Light Company (the licensee) requested amendments to Operating License Nos. DPR-31 and DPR-41 for Turkey Point Unit Nos. 3 and 4. The letters contained a LOCA analysis and proposed Technical Specifications changes in connection with operation of Units 3 and 4 with 28% of the steam generator plugged and a peaking factor of 2.125.

BACKGROUND

The Florida Power and Light Company submittal dated March 5, 1981 (ref. 1) provides a revised emergency core cooling system (ECCS) analysis for Turkey Point Units 3 and 4 with a steam generator tube plugging level of 28 percent. In addition to accounting for up to 28% overall tube plugging, the analysis utilizes new methods for the loss of coolant accident (LOCA) calculations and accounts for the new fuel rod models in NUREG-0630. This analysis supercedes the previous one submitted April 21, 1980 and ammended June 5, 1980 for a 25% level of steam generator tube plugging. Included in the 28% plugging level submittal are changes to the Technical Specifications and two non-LOCA design basis accidents judged to be affected by the increased steam generator plugging.

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The changes to the Technical Specifications requested by the licensee are the following:

a. Figure 2.1-1b

The curve has been updated to reflect the new steam generator tube plugging limit, and has been corrected per our NSSS vendor's recommendation.

b. Page 2.3-2

The overtemperature ΔT setpoint is now applicable for steam generator tube plugging $\leq$ 28 percent.

c. Page 2.3-3
The overpower ΔT setpoint values for > 19% and $\leq$ 25% are now applicable for steam generator tube plugging > 19% and $\leq$ 28%.

d. Page 3.1-7

Reactor coolant limits for >19% and $\leq$ 25% are now applicable for >19% and $\leq$ 28% steam generator tube plugging.

e. Page 3.2-3

The steam generator tube plugging limit in Specification 3.2.6a is increased to 28% and the F_Q to 2.125.

f. Figure 3.2-3

The K(Z) curve has been modified to reflect the new steam generator tube plugging limit and new F_Q.

The operating license for the Turkey Point power plant is being ammended to permit operation with up to 28% steam generator plugging and to take credit for an improved peaking factor, F_Q, resulting from new methods of LOCA analysis. The added steam generator tube plugging is necessary because of earlier corrosion problems which are not the subject of this review. It is expected that 1% to 2% of the tubes will need to be plugged in Unit 4 during the upcoming refueling outage which could put Unit 4 over the 25% plugging limit previously analyzed.

The effects of steam generator tube plugging are not as pronounced as might be expected considering that 28% of the heat transfer surface is inactive. The plugged tubes lead to a greater pressure drop across the steam generators and a 5% reduced loop flow. Despite the slightly reduced flow, the flow velocity in the remaining unplugged tubes is increased which improves heat transfer in the remaining tubes. Thus the reduction in heat transfer is less severe than the reduction in heat transfer surface. The reduced core flow (95% of FSAR value) necessitates a higher temperature increase across the core for the same power level. The average temperature has been kept the same, so the inlet temperature is lower and the outlet higher.

The LOCA analysis incorporates new methods not employed in previous analyses. In particular, the slip flow representation and downcomer region modelling are adopted from methods used in Upper Head Injection (UHI) plants. The methods have been approved for UHI plants, and applied to recent Zion and Millstone reloads. The effect of the model changes is to permit a larger peaking factor in spite of the increaed tube plugging.

The small break LOCA results are retained from previous analyses. Two Non-LOCA analyses, pump rotor seizure and control rod drive mechanism housing rupture accidents, are also addressed.

EVALUATION

The licensee has submitted an analysis of ECCS performance for both LOCA and non-LOCA conditions with steam generator tube plugging at the 28% level. The large break LOCA analysis was performed with $F_Q = 2.25$ and incorporated the fuel rod swelling and rupture models of NUREG-0630 which reduce F_Q to 2.125.

The non-LOCA analyses utilized an F_Q of 2.175 or higher to assure conservatism. Emphasis will be placed on the differences between this and previous submittals.

LOCA EVALUATION

The LOCA analysis was performed using a February 1978 version of the Westinghouse evaluation model (ref.2) with some modifications which are discussed in the next paragraph. The initial conditions and assumptions used in the large break analysis are the same with the exception of the peaking factor. Some of the input valves are:

Total Power	102% of 2200 MWt
Peaking Factor	2.25
Peak Linear Power	102% of 12.77 KW/ft
Accumulator Volume	875 ft^3 each

The most limiting break examined was a double ended cold leg guilotine break with a discharge coefficient of 0.4. A small break analysis was not performed since no significant change in results is expected from previous analyses.

UHI SOFTWARE TECHNOLOGY

In analyzing the large break LOCA, the applicant has proposed to use some of the modeling techniques currently approved for use in Westinghouse plants equipped with Upper Head Injection (UHI) (Reference 3). The following four changes were made to the SATAN VI computer program (Reference 4):

1) pseudo-viscosity
2) equation of state
3) modified drift flux
4) elevation pressure change

These changes were reviewed and approved for UHI plants in Reference 3. None of the four changes is unique to UHI plants and would be equally suitable to non-UHI plants. We therefore find those SATAN modifications acceptable for the Turkey Point large break analysis. The model approved in Reference 3 utilizes a split downcomer nodalization. This model was compared to several experimental results (Reference 4) and found acceptable. Since the experiments were not related to UHI, these comparisons would also be applicable to non-UHI plants such as Turkey Point.

In reference 5 Westinghouse analyzed a non-UHI plant using the UHI software technology discussed previously. Two calculations were done, one with a split downcomer nodalization and one with the traditional one-dimensional downcomer. The difference in PCT was only 11°F. Other comparisons of the split and one-dimensional downcomer models showed a similar small effect for non-UHI plants. Since the effect of the split downcomer model is small and the comparison to available data was reasonable we find the model acceptable.
To account more realistically for the actual Westinghouse 3-loop configuration, the intact loop nodalization was split back to steam generator. Although this was not done for UHI plants, it is actually a better representation for use with the split downcomer and is therefore acceptable. Addition of a containment node to better handle break flow slip is also acceptable. A sensitivity study to this change actually resulted in a slight increase in peak cladding temperature.

Several other non-substantive changes in SATAN not related to UHI technology are also acceptable.

ACCUMULATOR VOLUME SENSITIVITY

This sensitivity study is consistent with the requirements of Reference 6 and thus the analysis should be done with higher water volume.

CLADDING SWELLING AND RUPTURE

The NRC staff has been generically evaluating three materials models that are used in ECCS evaluations. Those models predict cladding rupture temperature, cladding burst strain, and fuel assembly flow blockages. We have (a) dicussed our evaluation with vendors and other industry representatives (Reference 7), (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis" (Reference 8), which concluded that licensing cladding models were in general, non-conservative, and (c) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG-0630 models were substituted for the present materials models in their ECCS evaluations (Reference 9 and 10).

Until we have completed our generic review and implemented new acceptance criteria for cladding models, we have required that the ECCS analyses be accompanied by supplemental calculations to be performed with the materials models of NUREG-0630. For these supplemental calculations only, we have accepted other compensatory model changes allowed for the confirmatory operating reactor calculations mentioned above. By letter dated March 5, 1981 (Ref. 1), the licensee provided a supplemental ECCS calculation. This calculation also accounted for a non-conservatism identified (Ref.11) by Westinghouse in their February, 1978 ECCS evaluation model, which used a fast-heatup-rate correlation for slow transients. Specifically, plant heatup rates are at slow temperature-ramp rates; whereas, the evaluation model was, in part, based on cladding tests that were conducted at fast temperature-ramp rates. The Turkey Point submittal assessed the combined impact of this calculational error and the final NUREG-0630 models to be worth 142°F peak cladding temperature above that previously calculated. Subsequently Westinghouse calculated that a reduction in total peaking factor F_Q of 0.125 would offset the portion of the

142°F increase in peak cladding temperature that exceeded 2200°F. Consequently an F_Q reduction is required for Turkey Point, and the licensee has amended the Technical Specifications to reflect a new F_Q of 2.125. We therfore conclude that the applicant has satisfied our concerns related to the swelling and rupture issue.

CONCLUSIONS

The changes in SATAN modeling techniques based on "UHI Technology" are acceptable for Large break analysis of the ECCS on Turkey Point Units 3&4 as described in this SER. They also meet the requirements of Appendix K to 10 CFR 50. The accumulator volume sensitivity and the resulting worst case determination is also acceptable. The results of the large break analysis with F_Q = 2.25 indicate a peak cladding temperature of 2183°F, maximum local cladding oxidation of 7.39%, and an overall cladding oxidation of less than 0.3%. All of these values are below the limits specified in 10 CFR 50.46.

The fuel rod swelling and rupture models in NUREG-0630 were examined and their impact suitably assessed. This assessment showed that a reduction of F_Q by 0.125 was required for a non-burst node to meet the LOCA acceptance criteria. Thus, the new F_Q is 2.25 - .125 = 2.125 to meet acceptance criteria.

Florida Power and Light has evaluated the large break LOCA accident with 28% steam generator tube plugging and consideration of NUREG-0630 models. The changes to the LOCA evaluation model appear acceptable and allow F_Q to be increased to 2.125. The previous valve of F_Q was 1.93 with 25% steam generator tube plugging (Ref.12). Because the new methods have been justified, the increased tube plugging level and F_Q are acceptable and meet current criteria for ECCS performance.

NON-LOCA ANALYSES

The non-LOCA accidents and transients are affected in a variety of ways by increased steam generator tube plugging. The excess heat removal accidents tend to be slightly less severe because of the impaired heat transfer. Other accidents, such as overpressurization events are essentially the same. This review concentrates on those events judged to be adversely impacted by increased steam generator plugging.

Three different accidents were reviewed in a safety evaluation report at the 25% steam generator tube plugging level (ref.13). Examination of an uncontrolled control rod assembly withdrawal at power showed that the more stringent thermal and hydraulic safety limits increase the margin to DNB. For 28% plugging, the safety limits are still applicable, so the results should remain acceptable. The loss of reactor coolant flow transient considered the simultaneous loss of electrical power to all coolant pumps. Tube plugging causes a quicker pump coastdown due to the increased pressure drop. The DNBR at 25% tube plugging was 1.48, which allows adequate margin (at 28% plugging) to keep the DNBR above 1.30. The last non-LOCA accident considered in reference 13 was the boron dilution accident. In this the dilution times are ample to cover for the slightly reduced volume for 28% tube plugging.

Two other non-LOCA accidents are analyzed in the current submittal (ref. 1) at the 28% steam generator tube plugging level. Both have $F_Q = 2.175$ to allow a small extra margin over the LOCA limit of 2.125. The locked rotor accident is more severe than the loss of reactor flow accident discussed earlier. The locked rotor analysis is based on a hot spot heat transfer calculation with F_Q = 2.55 and 100% flow.The latest submittal (ref. 1) states that F_Q = 2.175 yields a 14% benefit (by decreased energy input to the hot spot) while the 95% coolant flow causes a 5% reduction in benefits. Although this argument is not rigorous, the margins are sufficient to not require further analysis of the locked rotor accident at 28% plugging.

The rupture of a control rod drive mechanism housing is also examined in reference 1. The previous analysis was performed with F_Q = 2.32 and 100% flow. In this case the beneficial margin due to a lower F_Q cannot adequately compensate for the detrimental effects of decreased flow. However, there is an ample 400°F margin to the peak allowable temperature of 2700°F for this accident. This margin is sufficient to give confidence that the peak allowable temperature limit for this accident will not be exceeded.

SUMMARY

Based on the review of the submitted documents, we conclude that the results of the LOCA analysis with F_Q = 2.125 meet the criteria of 10 CFR 50.46. In addition the non-LOCA analyses submitted are conservative relative to the appropriate Standard Review Plan criteria. We conclude that the changes to the Technical Specifications for Turkey Point Units 3 and 4 are acceptable for up to 28% tube plugging.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types of 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 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 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: June 23, 1981

REFERENCES

1. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC) dated March 5, 1981, serial # L-81-99

2. WCAP-9220, Westinghouse ECCS Evaluation Model, February 1978 version, February 1978.

3. NUREG-0297 Safety Evaluation Report on Westinghouse ECCS Evaluation Model for Plants with Upper Head Injection, April, 1978

4. M. Y. Young, et al., "Westinghouse Emergency Core Cooling System Evaluation Model Application to Plants Equipped with Upper Head Injection," WCAP-8479-P, Revision 2, November 1977, Addendum 1, October 1979.

5. NS-TMA-2448 Letter from T. M. Anderson (Westinghouse) to J. R. Miller (NRC) dated May 15, 1981.

6. John F. Stolz, NRC, Letter to T. M. Anderson, Westinghouse, "Safety Evaluation of WCAP-9220, Westinghouse ECCS Evaluation Model, February 1978 Version," dated September 5, 1978.

7. Memorandum from R. P. Denise, NRC, to R. J. Mattson, "Summary Minutes of Meeting on Cladding Rupture Temperature, Cladding Strain, and Assembly Flow Blockage," November 20, 1979. Available in NRC PDR for inspection and copying for a fee.

8. D. A. Powers and R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NRC Report NUREG-0630, April 1980. Available from the NRC Division of Technical Information and Docket Control.

9. Letter from D. G. Eisenhut, NRC, to All Operating Light Water Reactors, dated November 9, 1979. Available in NRC PDR for inspection and copying for a fee.

10. Memorandum from H. R. Denton, NRC, to Commissioners, "Potential Deficiencies in ECCS Evaluation Models," November 26, 1979. Available in NRC PDR for inspection and copying for a fee.

11. Letter from T. M. Anderson, Westinghouse Electric Corporation, to D. G. Eisenhut, NRC, Number NS-TMA-2163, dated November 16, 1979.

12. Letter from S. A. Varga (NRC) to R. E. Uhrig (FPL) dated May 15, 1980

13. Letter from S. A. Varga (NRC) to R. E. Uhrig (FPL), dated October 26, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-250 AND 50-251

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 68 to Facility Operating License No. DPR-31, and Amendment No. 60 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 incorporate the results of a revised ECCS analysis for a steam generator plugging limit of 28%.

The application for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. 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 March 5, 1981, (2) Amendment Nos. 68 and 60 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 Library, Florida International University, Miami, Elorida 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 23 day of June, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
Figure 2.1-16	Figure 2.1-16
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 3.2-3	Figure 3.2-3

REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS,
THREE LOOP OPERATION

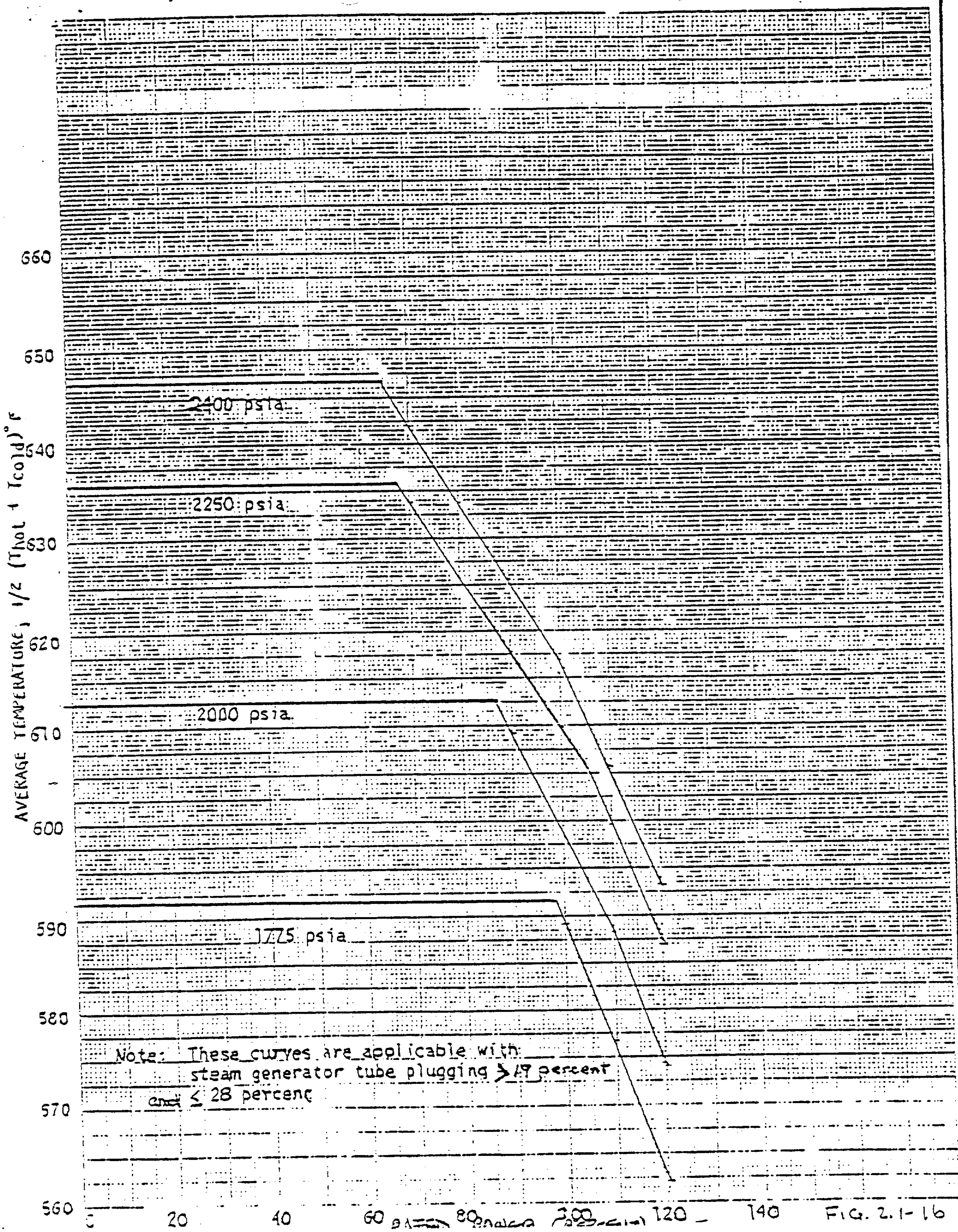

FIG. 2.1-16