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ACRS (16)
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JAN 27 1978

Docket No. 50-250

Florida Power and Light Company
ATTN: Dr. Robert E. Uhrig
Vice President
P. O. Box 013100
Miami, Florida 33101

Gentlemen:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-31 for the Turkey Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application dated June 8, 1977, supplemented by letter dated July 11, 1977.

This amendment authorizes operation of Turkey Point Unit No. 3 with up to an average of 15% of the tubes in the three steam generators in a plugged condition.

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

Original signed by
S. Nowicki for
George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No.
2. Safety Evaluation
3. FEDERAL REGISTER Notice

cc w/enclosures:
See page 2

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	CParrish	RClark:mjf	S Goldberg	GLear		
DATE	1/23/78	1/23/78	1/27/78	1/27/78		

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

January 27, 1978

Docket No. 50-250

Florida Power and Light Company
ATTN: Dr. Robert E. Uhrig
Vice President
P. O. Box 013100
Miami, Florida 33101

Gentlemen:

The Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. DPR-31 for the Turkey Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application dated June 8, 1977, supplemented by letter dated July 11, 1977.

This amendment authorizes operation of Turkey Point Unit No. 3 with up to an average of 15% of the tubes in the three steam generators in a plugged condition.

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

Stanley Norris

for George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 31
2. Safety Evaluation
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cc w/enclosures:
See page 2

cc:

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Honorable Dewey Knight
County Manager of Metropolitan
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Florida International University
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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. 31
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 June 8, 1977, 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. 31, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 27, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 31

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 pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Replace
3.2-3	3.2-3
B3.2-4	B3.2-4
B3.2-6	B3.2-6

reactivity insertion upon ejection greater than 0.3% Δ k/k at rated power. Inoperable rod worth shall be determined within 4 weeks.

b. A control rod shall be considered inoperable if
   (a) the rod cannot be moved by the CRDM, or
   (b) the rod is misaligned from its bank by more than 15 inches, or
   (c) the rod drop time is not met.

c. If a control rod cannot be moved by the drive mechanism, shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the inoperable rod.

5. CONTROL ROD POSITION INDICATION

If either the power range channel deviation alarm or the rod deviation monitor alarm are not operable rod positions shall be logged once per shift and after a load change greater than 10% of rated power. If both alarms are inoperable for two hours or more, the nuclear overpower trip shall be reset to 93% of rated power.

6. POWER DISTRIBUTION LIMITS

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

$F_Q(Z) \leq (2.22/P)$*x $K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.44)$*x $K(Z)$ 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 and Z is the core height location of F_Q.

b. Following initial loading before the reactor is operated above 75% of rated power and at regular effective full rated power monthly intervals thereafter, power distribution maps, using the movable detector system shall be made, to conform that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

*For tube plugging in excess of 10%, these values become (2.20/P) and (4.40) respectively.

An upper bound envelope of 2.22*times the normalized peaking factor axial dependence of Figure 3.2-3 has been determined to be consistent with the technical specifications on power distribution control as given in Section 3.2.

When an F_q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate experimental uncertainty allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_q, (b) 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 predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of start-up physics tests, at least once each full rated power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear

*For steam generator tube plugging in excess of 10%, this value becomes 2.20.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-250

Introduction

By letter dated June 8, 1977 (L-77-172), supplemented by letter dated July 11, 1977 (L-77-217) Florida Power and Light Company (the licensee) submitted a reevaluation of the Emergency Core Cooling System (ECCS) performance for Turkey Point Units Nos. 3 and 4. The reevaluation was performed for 15 percent of the steam generator tubes plugged, using the assumptions employed in the previous ECCS reevaluation (References 1 and 2) which has been reviewed and approved by the staff. A request was also made by the licensee to decrease the value of the peaking factor, F_q, in the Technical Specifications from 2.22 to 2.20 whenever steam generator tube plugging exceeds 10 percent. The reason for this reevaluation is to allow the plants to be operated with more than 10 percent of the steam generator tubes plugged. Turkey Point Unit No. 4 was authorized to operate with up to an average of 15% of the steam generator tubes in a plugged condition by our letter of August 3, 1977. This amendment will similarly authorize Unit No. 3 to operate with up to an average of 15% of the tubes plugged in the three steam generators.

Discussion

As noted above, the licensee's submittals of June 7, 1977 and July 11, 1977 included a reevaluation of the ECCS performance for both Turkey Point Units Nos. 3 and 4. After staff review and evaluation, an Order was issued on August 3, 1977 which, among other restrictions, changed the Heat Flux Hot Channel Factor (F_q) on page 3.2-3 of the Unit No. 4 Technical Specifications from 2.22 to 2.20. With this restriction, the staff concluded that Turkey Point Unit No. 4 could be safety operated with up to 15% of the steam generator tubes plugged.

On November 23, 1977, Unit No. 3 was shutdown for refueling and for inspection of the steam generators. Prior to the shutdown, about 5.7% of the tubes in the Unit No. 3 steam generators had been plugged. As a result of the steam generator inspection, an additional 5.7% of the tubes have been plugged so that at present an average of 11.4% of the total tubes in the Unit No. 3 steam generators are plugged. The tubes plugged were primarily those where the tube wall had thinned significantly

due to the tube denting phenomena. Since more than 10% - but less than 15% - of the tubes in the steam generators are plugged in Unit No. 3 a reduced heat flux hot channel factor (F_q) limit is also required for Unit No. 3 as a result of the reevaluation of the ECCS analysis. Thus, with this amendment, both Units Nos. 3 and 4 will be operated with the same restriction on F_q and the three pages of the Technical Specifications being changed by this amendment for Unit No. 3 will be the same as for Unit No. 4.

Evaluation

ECCS Performance Evaluation

The ECCS analysis provided by Florida Power and Light Company for Turkey Point Units Nos. 3 and 4 consisted of an evaluation of ECCS performance using the October 1975 version of the Westinghouse ECCS evaluation model and included the following assumptions:

(a) 15 percent steam generator tubes plugged
(b) Maximum Peaking Factor, $F_q = 2.20$
(c) Accumulator minimum water volume: 875 ft^3
(d) Upper head fluid temperature equal to reactor vessel outlet (hot leg) temperature
(e) Coolant inlet temperature, $T_{in} = 550.2°F$ (nominal + 4°F).

The analysis was performed for a double ended cold leg guillotine break (DECLG) with discharge coefficient, $C_d = 0.4$. This break size was identified as the critical break in the previous evaluation (References 1 and 2) performed using the same assumptions except for steam generator tube plugging which was assumed to be 10 percent. The licensee has demonstrated in the submittal of July 11, 1977 that the critical break size will remain unchanged for 15 percent of the steam generator tubes plugged.

The results of the analysis indicate that the limiting values for peak clad temperature and local Zr - water reaction are: 2173°F and 11.655 percent, respectively. Both these values are below the limits specified in 10 CFR 50.46.

The sensitivity study performed in the previous ECCS analyses considered the effect of the degree of steam generator tube plugging on peak clad temperature and local Zr - water reaction. It was demonstrated that both these parameters increase with increasing percentage of plugged tubes. Therefore, the applicability of the present analysis is limited to operation of Turkey Point Units Nos. 3 and 4 with up to 15 percent plugged tubes. Similar restriction would apply to the maximum peaking factor (F_q). For 10 percent or less of the steam generator tubes plugged $F_q = 2.22$ and for plugged values between 10 and 15 percent $F_q = 2.20$. Whenever the plugging exceeds 15 percent a new ECCS analysis would have to be performed.

Recently the NRC staff has noted that in LOCA calculations for some PWRs, a decrease in the temperature of primary coolant at the reactor vessel inlet has resulted in a predicted increase in peak clad temperature. In discussions with the PWR vendors the NRC staff has learned that they have all observed this trend while performing LOCA calculations with their individual approved evaluation models. In the past, it has been widely accepted that it was conservative to assume the highest possible initial inlet coolant temperature for LOCA calculations (typically maximum full power operating temperature +4° for measurement uncertainty). The apparent cause of this behavior stems from the fact that a reduction in coolant inlet temperature results in a reduction in the coolant saturation pressure. This decreases the flow rate from the vessel side of the break after the short period of subcooled blowdown. This reduced flow, for the postulated cold leg break, decreases the magnitude of the downward flow rate through the core that exists for a large portion of the blowdown period. This decreases the heat transfer coefficient and consequently less stored energy is removed during blowdown.

Reducing the coolant inlet temperature also changes the flow rate from the top of the vessel to the hot leg and out of the break through the steam generator and reactor coolant pump. The changes in hot leg flow caused by a reduction in inlet temperature tend to decrease the core flow rate during the period of positive core flow. This also leads to the removal of less stored energy during blowdown. Thus, the fuel temperature is higher at the end of bypass. Most PWRs exhibit peak clad temperature during reflood, and entering the reflood period with a greater fraction of stored heat remaining after blowdown may cause an increase in the peak clad temperature. The decreased negative core flow may extend the time to end of bypass. In the evaluation model, more accumulator water is assumed to spill out of the break. If, as a result of this assumption, there is insufficient accumulator water remaining to fill the downcomer, reflood will be delayed. This will also contribute to the increase in peak clad temperature.

However, a reduction in coolant inlet temperature may not always result in an increase in peak clad temperature. It has been observed that if the clad rupture location changes to a different elevation where the core power is less, peak clad temperature may decrease.

In addition to the predicted changes in blowdown core flow and heat transfer, reducing coolant inlet temperature also causes a slight reduction in containment back pressure during reflood. Reducing this pressure is known to result in lower reflood rates with correspondingly higher clad temperatures. However, the effect due to containment back pressure is minor compared to blowdown core flow and heat transfer effects.

At this time the NRC staff believes that nominal values of inlet temperatures and steam generator shell side steam conditions should be used in all LOCA calculations since the effects of variations in inlet temperatures

and steam conditions on peak clad temperatures are not consistent and are at best second order effects. The NRC staff is currently seeking additional information to be used in a generic evaluation of the effect of coolant inlet temperature and steam conditions on ECCS performance.

The maximum sensitivity of peak clad temperature to inlet temperature that the NRC staff has seen to date shows that for a 1°F decrease in primary coolant inlet temperature the peak clad temperature following a large break LOCA would increase 4°F. However, a reduction in inlet temperature results in a corresponding reduction in core average temperature and steam generator shell side steam pressure. A reduction in steam generator secondary pressure results in lower peak clad temperatures. If we now assume a decrease in coolant inlet temperature and correspondingly adjust the steam generator shell side steam conditions, the peak clad temperature for Turkey Point would increase less than 20°F. We consider this letter increase to be a minor second order effect. Thus, we conclude that the current ECCS analysis on file for Turkey Point meets the criteria of 10 CFR 50.46 and is, therefore, acceptable.

Environmental Consideration

We have determined that the amendment does 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 amendment involves 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 amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 27, 1978

References

1. Letter from Robert E. Uhrig, FPL, to NRC, L-76-419, dated December 9, 1976, transmitting "Major Reactor Coolant System Pipe Rupture Analysis."

2. Letter from Robert E. Uhrig, FPL, to NRC, L-77-1, dated January 3, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-250

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 31 to Facility Operating License No. DPR-31, issued to Florida Power and Light Company, which revised Technical Specifications for Operation of the Turkey Point Nuclear Generating Unit No. 3, located in Dade County, Florida. The amendment is effective as of the date of issuance.

The amendment authorizes operation of Turkey Point Unit No. 3 with up to an average of 15% of the steam generator tubes in a plugged condition.

The application for the amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated June 8, 1977 (as supplemented by letter dated July 11, 1977), (2) Amendment No. 31 to License No. DPR-31, 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 27th day of January 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Stanley J. Nowicki, Acting Chief
Operating Reactors Branch #3
Division of Operating Reactors