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Docket Nos. 50-250
and 50-251

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

Gentlemen:

By our letter dated September 22, 1978, we transmitted to you Amendment Nos. 38 and 31 to Facility License Nos. DPR-31 and DPR-41 for Turkey Point Unit Nos. 3 and 4.

The amendment number on the Amendment, Technical Specifications pages, Safety Evaluation, and the Notice for Unit No. 3 was incorrectly stated as Amendment No. 37. The September 22, 1978 transmittal letter correctly stated the Unit No. 3 amendment number as No. 38.

For ease in correcting this error we are reissuing the September 22, 1978 Amendment Nos. 38 and 31 in their entirety.

Sincerely,

ORIGINAL SIGNED

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

Enclosure:
Corrected Amendment Nos. 38 and 31
dated September 22, 1978

cc: w/enclosure
See next page

Construct
ccp

OFFICE	DOR:ORB1	DOR:ORB1	DOR:ORB1			
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FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
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 19, 1978, supplemented on July 10 and 20, 1978, August 9 and 16, and September 13, 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 the 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. 38 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment supercedes the Order for Modification of License dated June 7, 1978 and is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer / for

Darrell G. Eisenhut, Assistant Director
for Systems and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 22, 1978

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING STATION UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
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 June 19, 1978, supplemented on July 10 and 20, August 9 and 16, and September 13, 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 paragraphs 3.B and 3.D of the Facility Operating License No. DPR-41 are hereby amended to read as follows:

3.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.D Steam Generator Operation

1. Turkey Point Unit 4 shall be brought to the cold shutdown condition in order to perform an inspection of the steam generators after six equivalent months of Cycle 5 operation from September 22, 1978. Nuclear Regulatory Commission (NRC) approval shall be obtained before resuming power operation following this inspection. For the purpose of this requirement, equivalent operation is defined as operation with a reactor coolant temperature greater than 350°F.

2. Reactor coolant to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm per steam generator. With an steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours. The leaking tube(s) shall be evaluated and plugged prior to resuming power operation.

3. The concentration of radioiodine in the reactor coolant shall be limited to 1.0 microcurie/gram during normal operation and to 30 microcuries/gram during power transients.

4. Reactor operation shall be terminated and NRC approval shall be obtained prior to resuming operation if primary to secondary leakage attributable to the denting phenomena is detected in 2 or more tubes during any 20 day period.

5. The Metal Impact Monitoring System (MIMS) shall be continued in operation with the capability of detecting loose objects. If the MIMS is out of service in other than cold shutdown or refueling mode of operation, this fact shall be reported to the NRC. Any abnormal indications from the MIMS shall also be reported to the NRC by telephone by the next working day and by a written evaluation within two weeks.

6. Following each startup from below 350^{o}F, core barrel movement shall be evaluated using neutron noise techniques.

4. This license amendment supercedes the Orders for Modification of License dated August 3 and 11, 1977 and March 8 and June 7, 1978 and is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

O. Schwencer / for

Darrell G. Eisenhut, Assistant Director
for Systems and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 22, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 38

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-31

DOCKET NO. 50-250

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

Remove	Replace
2.3-2	2.3-2
2.3-3	2.3-3
3.2-3	3.2-3
Figure 3.2-3	Figure 3.2-3
3.1-7	3.1-7
B3.2-4	B3.2-4
B3.2-6	B3.2-6

Add Figure 2.1-1b

ATTACHMENT TO LICENSE AMENDMENT NO. 31

TO THE TECHNICAL SPECIFICATIONS

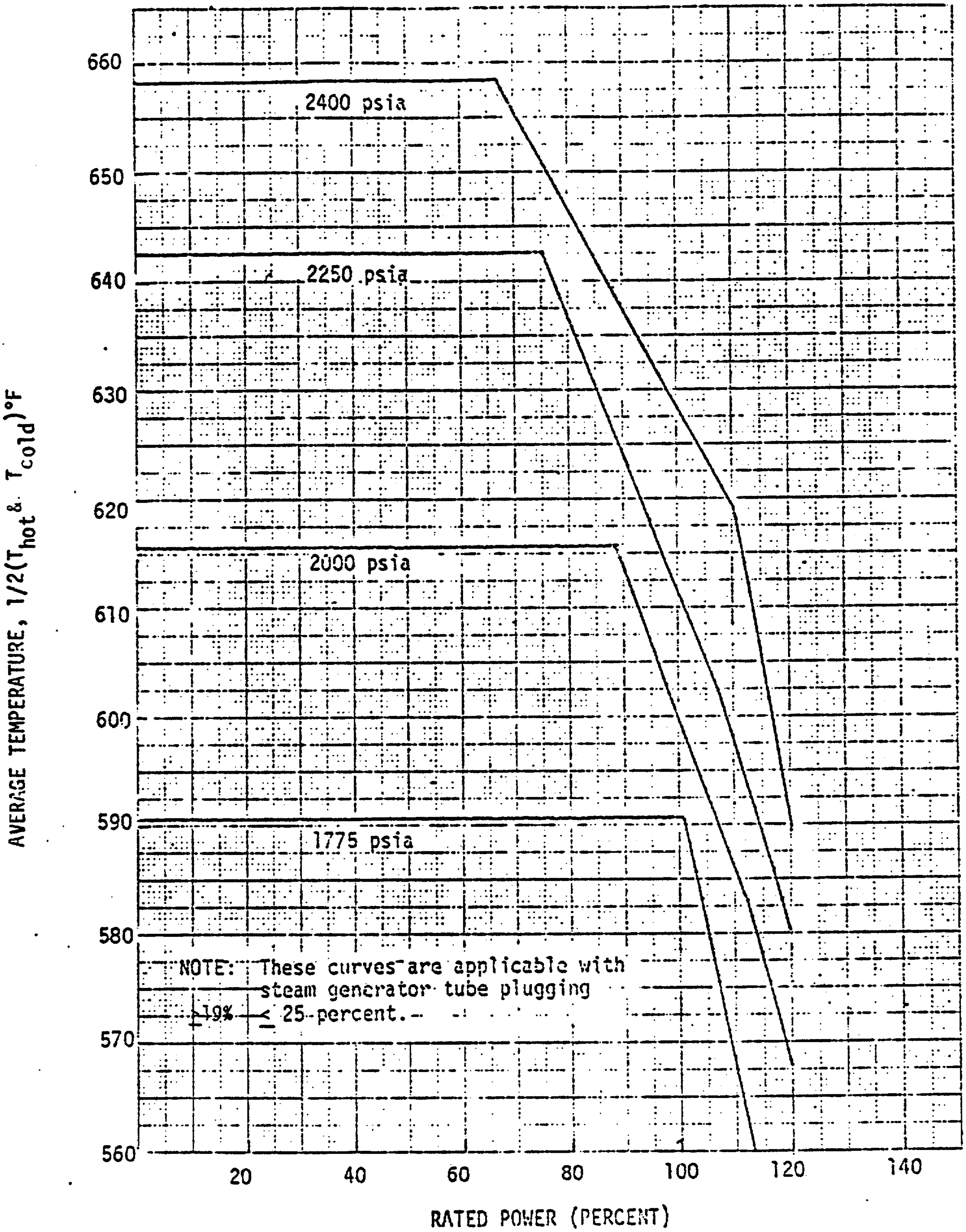
FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-251

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

Remove	Replace
2.3-2	2.3-2
2.3-3	2.3-3
3.2-3	3.2-3
Figure 3.2-3	Figure 3.2-3
3.1-7	3.1-7
B3.2-4	B3.2-4
B3.2-6	B3.2-6

Add Figure 2.1-1b



REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS,
THREE LOOP OPERATION

Figure TS 2.1-1b
Amendment Nos. 38 & 31

Reactor Coolant Temperature

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

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 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 $\leq$ 25 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 %

This factor is 1.08 for steam generator tube plugging >19% and $\leq$25%.

+This factor is 0.00106 for steam generator tube plugging >19% and $\leq$25%.

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 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. <u>CONTROL ROD POSITION INDICATION</u>

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. <u>POWER DISTRIBUTION LIMITS</u>

a. Hot channel factors:

With steam generator tube plugging $\leq$ 25%, the hot channel factors (defined in the basis) must meet the following limits at all times except during low power physics tests:

$$F_q (Z) \leq (2.03/P) \times K(Z), \text{ for } P > .5$$

$$F_q (Z) \leq (4.06) \times K(Z), \text{ for } P \leq .5$$

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

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 predicted F_q exceeds 2.03, the power will be limited to the rated power multiplied by the ratio of 2.03 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 confirm that the hot channel factor limits of the specification are satisfied. For the purpose of this comparison,

HOT CHANNEL FACTOR-NORMALIZED
OPERATING ENVELOPE (FOR STEAM
GENERATOR TUBE PLUGGING $\leq$25% and F_q=2.03)

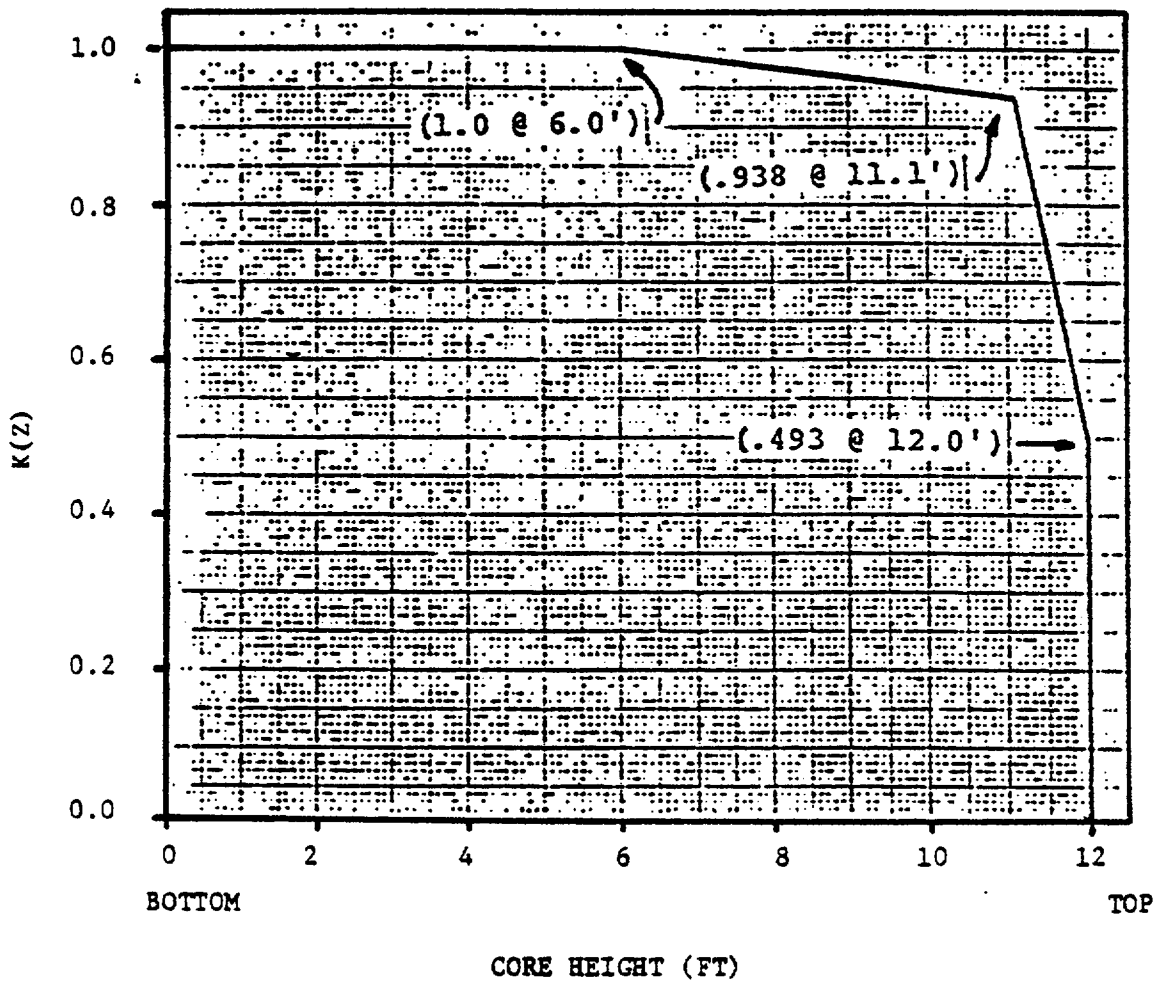


FIGURE 3.2-3

6. **DNB PARAMETERS**

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

a. Reactor Coolant System Tavg ≤ 578.2°F

b. Pressurizer Pressure ≥ 2220 psia*

c. Reactor Coolant Flow ≥ 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 ≥ 268,500 gpm for steam generator tube plugging ≤ 15%.

Reactor Coolant Flow ≥ 263,130 gpm for steam generator tube plugging > 15% and ≤ 19%.

Reactor Coolant Flow ≥ 255,075 gpm for steam generator tube plugging > 19% and ≤ 25%.

An upper bound envelope as defined by 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

Flux Difference ($\Delta\phi$) and a reference value which corresponds to the full design power equilibrium value of Axial Offset (Axial Offset = $\Delta\phi$/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the F_q upper bound envelope 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 the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal rated power operating position appropriate for the time in life. Control rods are usually withdrawn farther as burnup proceeds). This value, divided by the fraction of design power at which the core was operating is the design power value of the target flux difference. Values for all other core power levels are obtained by multiplying the design power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of $\pm 5\%$ ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every rated power month. For this reason, methods are permitted by Item 6c of Section 3.2 for updating the target flux differences. Figure B3.2-1 shows a typical construction of the target flux difference band at BOL and Figure B3.2-2 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during the required, periodic excore calibra-

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 38 AND 31 TO LICENSE NOS. DPR-31 AND DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

Introduction

By application dated June 19, 1978 and supplemented on July 10 and 20, August 9 and 16 and September 13, 1978 (1, 2, 3, 4, 5, 16)*, the Florida Power and Light Company (the licensee) requested amendments to Operating License Nos. DPR-31 and DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4. The application, which contains accident analyses and proposed Technical Specification changes is in support of a request to modify the Technical Specifications in connection with the refueling of Unit No. 4 for Cycle 5 operation and the operation of Unit Nos. 3 and 4 with up to an average of 25% of the tubes in the three steam generators in each unit in a plugged condition. The application also responds to the Order for Modification of the Licenses dated June 7, 1978 (17). That order required that FPL submit a reevaluation of the ECCS cooling performance corrected for certain errors in the zirconium water reaction.

In addition, the steam generator inspection report for Turkey Point Unit No. 4 required by the Orders for Modification of License dated August 3 and 11, 1977 (18) and March 8, 1978 (19) has been submitted for NRC review and approval.

Turkey Point Unit No. 4 has been reloaded for Cycle 5 operation and is expected to be ready for restart on or about September 22, 1978. There are no changes in fuel or in the Technical Specifications brought about directly by this reload. However, NRC Orders (18, 19) require a steam generator tube inspection which must be approved by the NRC before the reactor may be returned to operation. An early estimate of the number of steam generator tubes that might require plugging indicated that it might be necessary to plug more than the 19% allowed by current Technical Specifications. Consequently, FPL

*References are indicated by numbers in parenthesis and may be found at the end of this safety evaluation.

requested permission to plug up to 25% of the steam generator tubes in each unit.

The NRC requirements for approval to operate with plugged steam generator tubes include an ECCS reevaluation. The NRC Order of June 7, 1978 (17) required that "as soon as possible, the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with the Westinghouse Evaluation Model, approved by the staff and corrected for the errors described within.". Consequently, since the model had been corrected for the errors and we had approved that correction (11) the FPL application (4) for permission to plug up to 25% of the steam generator tubes also requested that the provisions of the June 7, 1978 Order (12) be deleted.

Our Orders for Modification of the License dated August 3 and 11, 1977 (18) and March 8, 1978 (19) placed limitations on the operation of Turkey Point Unit No. 4 in relation to steam generator tubes. These limitations are being retained in the license by this amendment and the Orders are thus removed. The basis for this change is that past experience and the review of the latest inspection of the steam generators with plugged tubes has shown that the required margin of safety is being retained by the licensee.

Following is our evaluation for the action discussed above which provides the basis for concluding that the Turkey Point Unit No. 4 can be safely returned to operation upon completion of the current steam generator plugging and refueling operation.

I. RELOAD UNIT 4 CYCLE 5 AND 25% STEAM GENERATOR TUBE PLUGGING - UNIT 3 AND 4

Discussion

By the application dated June 19, 1978[1], as supplemented July 10, 1978,[2] July 20, 1978[3], August 9, and 16, 1978 [16] and September 13, 1978.[5] Florida Power and Light Company (the licensee) proposed to change the Technical Specifications for the Turkey Point Units Nos. 3 and 4 in connection with the refueling of Unit 4 for Cycle 5 operation. The first reference concerns reloading Unit 4 only. It states that subsequent submittals will contain license amendment requests to allow full power operation with 25% steam generator tubes plugged and to incorporate ECCS model changes. The current Turkey Point 3 and 4 safety analyses are valid for steam generator tube plugging levels of up to 19%. The proposed license amendment to allow operation with 25% steam generator plugging is contained in references 2 , 3 and 16. The ECCS model changes are discussed in references 4 and 5.

The refueling consists of the replacement of 61 burned fuel assemblies by 12 fresh assemblies and 49 previously burned assemblies. The previously burned assemblies are: 24 assemblies last irradiated in Cycle 2 with an approximate average burnup of 25,000 MWD/MTU and 25 assemblies last irradiated in Cycle 3 with an approximate average burnup of 27,700 MWD/MTU. Use of a limited number of fresh assemblies and a large number of assemblies with high burnup will make Cycle 5 a short cycle of approximately six months duration. The licensee has elected to pursue this course of action to provide for contingencies in possible steam generator replacement.

In order to flatten the radial power distribution the licensee will place 8 fresh borosilicate burnable poison rods in each of four centrally located once burned fuel assemblies, and 12 depleted borosilicate burnable poison rods in each of 28 fuel assemblies, spaced throughout the core.

Analyses performed for the Cycle 5 reload core design were based on the following assumptions:

1. Cycle 4 operation is terminated after 9400^{+200}_{-100} MWD/MTU
2. Cycle 5 burnup is limited to the end of full power capability, and
3. There is adherence to plant operating limitations given in the Technical Specifications.

The licensee has proposed the following changes to the Technical Specifications for both Units 3 and 4 as a result of its analyses of operation with 25% steam generator tube plugging and the LOCA:

1. Add a figure defining the safety limits for 3 loop operation with between 19 and 25 percent of the steam generator tubes plugged.

2. Change the overpower ΔT trip function constants consistent with the above safety limits.

3. Reduce the reactor coolant flow rate to 255,075 gpm (95% of former value).

4. Change the total core peaking factor, F_Q to 2.03.

5. Revise the axial F_Q shaping factor figure to reflect a renormalization of the new (large break) LOCA analysis with the existing function.

Evaluation

Fuel Mechanical Design

The mechanical design of the fresh fuel assemblies (Region 7) is identical to the Region 6 fuel loaded in the last core reload.

Clad flattening will not occur during Cycle 5. Clad flattening time is predicted to be greater than 34,000 EFPH for all fuel regions being irradiated during Cycle 5 using the approved Westinghouse Evaluation Model[(6)]. Since the maximum cumulative irradiation time through Cycle 5 for the limiting region (Region 3) is expected to be approximately 25,600 EFPH, clad flattening will not occur.

Control Rod Insertion Limits

There are no changes proposed to the control rod insertion limits for Cycle 5. There are a number of criteria which the control rod insertion limits are checked against each cycle. The most important of these are shutdown margin, ejected control rod worth, and $F_{\Delta H}$. The existing insertion limits remain adequate to meet the control requirements for Cycle 5.

Shutdown Margin

The hot full power shutdown margin is predicted by the licensee to be 3.23% $\Delta\rho$ at BOC and 2.69% $\Delta\rho$ at EOC, compared to a shutdown margin requirement of 1.36% $\Delta\rho$ at BOL* and 1.77% $\Delta\rho$ at EOL as assumed in the steam line break

*The normal BOL requirement is 1% shutdown margin. Because of the short Cycle 5 life, the initial boron concentration will be low enough to require a 1.36% shutdown margin.

analysis. This is acceptable because of extra margin between predicted and required shutdown margin throughout cycle life. In addition, the predicted shutdown margin is conservative because a 10% calculational uncertainty is subtracted from the all rods inserted except for the highest worth stuck rod calculation in determining the predicted shutdown margin. Furthermore, confirmation of the validity of the prediction is made during the startup physics test program by measuring the regulating banks, which contain about half of the total control rod worth. These measured worths are compared with predictions for the measurement conditions made with the same model used for calculating the shutdown margin.

Reload Transient and Accident Analysis

The licensee has presented the results of Westinghouse predictions of the core kinetics parameters for Cycle 5. These are calculated with methods used and accepted for all recent reloads of Westinghouse designed reactors. The Cycle 5 kinetics parameters remain within the bounds of the limits found acceptable for previous cycles.

The licensee's evaluation of peaking factors for the rod out of position and dropped rod cluster control assembly (RCCA) incidents show that departure from nucleate boiling ratio (DNBR) is maintained above 1.30. For the dropped bank incident, the turbine runback is sufficient to present a DNBR less than 1.30. Since the DNBR remains above 1.30, the consequences of these incidents for Cycle 5 are acceptable.

The licensee evaluated the hypothetical steam line break cases with and without a loss of offsite power. The results of this evaluation indicated that a reanalysis of the hypothetical breaks inside containment without off-site power was required. The analysis used the same design methods and assumptions approved for previously submitted accident analyses except for the method of calculating the Doppler power coefficient. The Cycle 5 coefficient properly accounts for the effects of reduced reactor coolant flow which exists for the cases with loss of offsite power. This includes the effects of local density variations as a function of flow rate and power level. The transient results show that for all hypothetical steam line break cases the DNB acceptance criteria are met. The conclusions of the FSAR relative to meeting safety criteria remain valid and the results of this reanalysis are therefore acceptable.

Transient and accident analysis of both Unit 3 and Unit 4 with steam generator tube plugging up to 25% are considered in the following sections:

Reactor Coolant System Flow Rate

As the level of steam generator tube plugging increases the reactor coolant system (RCS) flow rate decreases. To quantitatively assess the effect of steam generator tube plugging on RCS loop flow, the licensee has taken measurements to obtain the loop flow rate at several levels of steam generator tube plugging.

The data points were compared with the flow rate predictions obtained with the Westinghouse analytical model. The maximum deviation between the measured and predicted curves was used as a constant bias to reduce the predicted curve of flow rate versus percent steam generator tubes plugged. This curve was then further reduced by 2% to account for measurement uncertainty, which the licensee has shown to be greater than the 2σ confidence limit on the measured flow rate.

The resulting curve indicates that at a plugging level of 25%, the flow rate will not be more than 5% below the thermal design flow rate of 89,500 gpm per loop. This flow rate, 85,025 gpm per loop, was then used in the evaluation of postulated transients and accidents for 25% steam generator tube plugging. The staff finds this acceptable.

Transients and Accidents

As a result of increasing the level of steam generator tube plugging to 25% three principal factors affect the assumptions used in the analyses of postulated transients and accidents. These factors are:

1. The RCS flow rate is lower than the thermal design value,

2. The RCS volume is less than that assumed in the reference analyses, and

3. The pump coastdown characteristics are more severe than those assumed in the reference analyses.

The licensee submitted an assessment of the impact of steam generator tube plugging up to a level of 25% on the non-LOCA incidents[2,3] for both Units 3 and 4. For each event the important parameters which were affected by the higher level of steam generator tube plugging were identified. Each event was then evaluated to determine how the impacted parameters affected the analysis. The evaluations were based on the following assumptions:

Parameter	This Analysis	Reference Analyses
Thermal design flow, gpm/loop	85,025	89,500
S. G. tube plugging, %	25	19 (0 FSAR)
*Power level, Mwt (100%)	2200	2200
*Tavg at 100% power, °F	574.2	574.2
ΔT at 100% power, °F	58.9	55.9
Steady state DNBR	1.72	1.8 (1.62 FSAR)
$F_{\Delta H}^{N}$	1.55	1.55 (1.75 FSAR)
F_Q maximum (non-LOCA)	2.05	2.55

*The analyses conservatively used 102% power (2244) and Tavg + 4° (578.2)

The results of the evaluation indicated that these events were limiting or most sensitive to the higher steam generator tube plugging level.

1. Uncontrolled Control Rod Assembly Withdrawal at Power

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. The increased steam generator tube plugging affects the analysis due to the reduction in RCS flow, the elevation in outlet temperature and the increase in loop transient time. As a result, the minimum departure from nucleate boiling ratio (DNBR) reported in the FSAR for fast reactivity insertion rates would be reduced by approximately 5%. However, FSAR analysis assumed an $F_{\Delta}{}^{N}{}_{H}$ of 1.75 versus the current limit 1.55. This would result in approximately 20% additional DNBR margin. Thus there would be a net increase in the minimum DNBR reported in the FSAR for fast reactivity insertions.

The overtemperature equation constants were recalculated consistent with the new Core Thermal and Hydraulic Safety Limits (T. S. Figure 2.1-1b) and compared to the FSAR values. The FSAR values were shown to be more limiting due to the higher $F_{\Delta}{}^{N}{}_{H}$ which was used for the original Core Thermal Limits. To offset the effects of the RCS flow reduction, the FSAR overtemperature ΔT trip constants will be maintained in the Technical Specifications (page 2.3-2). By using these same setpoints, the reduction in DNBR during the transient would be approximately the same. Thus the minimum DNBR for the 25% steam generator tube plugging case is expected to be greater than the FSAR value since the initial steady state DNBR has increased.

2. Loss of Reactor Coolant Flow

The most severe loss of flow transient is the simultaneous loss of electrical power to all three reactor coolant pumps. The increase in steam generator tube plugging affects the analysis due to increased loop resistance which results in a more rapid pump coastdown. This event was reanalyzed for 25% tube plugging and the resultant minimum DNBR is 1.48. Thus, adequate margin exists for the loss of flow event with the higher level of steam generator tube plugging.

3. Chemical and Volume Control System Malfunction

The analyses of boron dilution events are affected by increased steam generator tube plugging due to the reduction in RCS volume. The analysis of boron dilution during refueling will not be affected since for this case the volume of reactor coolant in the steam generators is not considered.

For dilution during startup and at power the reactor coolant volume in the steam generator tubes is assumed to be reduced by 25% (510 ft^3) due to the increased tube plugging. Thus the total RCS volume used in the analysis is reduced from 7800 ft^3 to 7290 ft^3. This results in approximately a 7% reduction in dilution time from startup conditions. The resultant 223 minutes for operator action is significantly greater than the acceptance criteria of 15 minutes.

For the dilution during power operation case the reactivity insertion rate versus boron concentration curve has been recalculated consistent with the reduced RCS volume. The results show that the reactivity insertion rate assumed in the FSAR is still valid. Thus the FSAR analysis of boron dilution during power operation is acceptable and the 15 minute acceptance criteria will be met for the higher level of steam generator tube plugging.

The staff has concluded, based on the results of the evaluations and analyses performed by the licensee, that the effects of the postulated transients and accidents are acceptable at steam generator tube plugging levels up to 25%.

ECCS Analysis

The licensee has provided[4,5,16] reanalysis of ECCS for both Units 3 and 4 using the recently modified Westinghouse evaluation model.[7,8,9,10] This model was recently reviewed and approved by the staff.[11] It includes the correction for the Zr-water error.

Presently, Turkey Point Station is operating with the interim values of total peaking factor of 2.02 and 1.97 for Units 3 and 4, respectively. These values were imposed by the Order for Modification of License[12] after an error in the heat generated by Zr-water reaction had been discovered in the Westinghouse ECCS evaluation model. The order requested the licensee to submit, as soon as possible, a reevaluation of the ECCS performance calculated in accordance with the corrected and approved evaluation model. The present submittal fulfills this requirement.

The licensee has evaluated the ECCS performance for a large break LOCA using the modified Westinghouse evaluation model and assuming 25 percent of steam generator tubes plugged. The analysis was performed for a double ended guillotine cold leg break (DECLG) with a discharge coefficient of C_D=0.4. The licensee has shown in the previous submittal[13] that this break size corresponds to the highest values of peak cladding temperature and Zr-water reaction. The licensee has also demonstrated that the break size remains unaffected by the amount of the steam generator tubes plugged.[14]

The input parameters assumed in the analysis are listed below:

Core Power: 102 percent of 2200 Mwt (rated power)
Peak Linear Power: 102 percent of 11.53 kw/ft
Peaking Factor: 2.03
Accumulator Water Volume: 875 ft^3 per accumulator.

The results of the ECCS analysis indicate a peak cladding temperature of 2173°F, a maximum local Zr-water reaction of 7.68 percent and a total Zr-water reaction of less than 0.3 percent. All these values are below the limits specified in 10 CFR 50.46.

The licensee did not include small break analysis since neither steam generator tube plugging nor correction of the Zr-water error affect significantly the results of this analysis.

The "18 case FAC analysis" was provided by the licensee(5) because the limiting peaking factor used in the analysis was below the value for which the excore detectors could give reliable measurements. The analysis showed that the maximum predicted F_Q that could occur for the remainder of Unit 3, Cycle 5 and for the upcoming Unit 4 Cycle 5 would never exceed the maximum allowable value derived from the corrected ECCS evaluation. The plant could therefore operate during Cycle 5 of both Units 3 and 4 without the augmented surveillance procedures which were discussed in reference 15.

Based on the review of the submitted documents, the staff concludes that the results of the ECCS reanalysis, performed with the revised February 1978 version of the Westinghouse ECCS evaluation model corrected for Zr-water reaction error and including the assumption of 25 percent steam generator tubes plugged, yield the values of LOCA parameters which are conservative relative to the 10 CFR 50.46 criteria. The staff considers the submitted ECCS reanalysis, for operation of the plants with up to a maximum of 25 percent steam generator tubes plugged, acceptable.

Technical Specifications

The licensee has proposed changes to the Technical Specifications to permit operation with up to 25% steam generator tube plugging for both Units 3 and 4.

Figure 2.1-1b, "Reactor Core Thermal and Hydraulic Safety Limits, 3 Loop Operation" has been added to the Technical Specifications for operation with between 19 and 25 percent of the steam generator tubes plugged. These limits were generated by Westinghouse and are consistant with the limits for lower levels of plugging.

The Overtemperature ΔT equation is unchanged for plugging up to 25%. The conservative FSAR constants will not be change as discussed above.

The Overpower ΔT equation constants were recalculated for 25% steam generator tube plugging. The values were more limiting for the reduced flow conditions and are therefore incorporated into the Technical Specifications.

The minimum reactor coolant flow has been reduced to 255,075 gpm for steam generator tube plugging between 19 and 25 percent. This is consistant with the flow assumed in the transient and accident analysis.

The Technical Specification for the maximum allowable full power value of F_Q is changed to 2.03. This is the value assumed as input for the LOCA analysis.

The normalized axial F_Q shaping factor (Figure 3.2-3) in the Technical Specifications has been changed consistent with the assumptions used as input for the LOCA analysis.

The staff has reviewed the proposed changes to the Technical Specifications and finds them consistent with the analyses discussed in the preceeding sections and therefore acceptable.

Startup Tests

The startup physics tests for Turkey Point Unit 4 Cycle 5 will verify nuclear design, power distribution and control rod worth predictions. This program includes low power critical boron concentration tests, temperature coefficient tests and rod worth and power distribution measurements. At higher powers, core power distribution and power coefficient tests will be performed. This program including the acceptance criteria for each test was reviewed by the staff. The program is described in reference 5.

The results of this startup physics test program will be submitted to the NRC in the form of a summary report within 45 days of completion of the program. The staff concludes that this program is acceptable.

Environmental Conclusions

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 issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that:

(1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amdnements will not be inimical to the common defense and security or to the health and safety of the public.

II. STEAM GENERATOR TUBE INSPECTION - UNIT 4

DISCUSSION

By letter dated September 6, 1978 (17), the licensee submitted the results of the steam generator tube inspection performed at Turkey Point Unit 4 during the August/September, 1978 refueling outage, including the plugging criteria applied to the three steam generators. Based on these inspection results, the implemented plugging patterns, and previously submitted ECCS analysis, FPL concludes that the facility can be returned to full power operation for at least six equivalent months.

Turkey Point Unit 4 has been operating under an August 3 and 11, 1977 (18) and March 8, 1978 (19), NRC Orders for Modification of Facility Operating License No. DPR-41. One of the conditions of these Orders was that NRC approval shall be obtained prior to resuming power operation following the mandated inspection of the steam generators.

EVALUATION

Inspection Program

The steam generator tube inspection performed during this shutdown included programs to assess the conditions associated with both the denting and "wastage" problems. For denting tube gauging was done in all three steam generators in order to assess the extent and pattern of tube denting. On the hot leg side, all tubes near the tube lane which were predicted to be bounded by the 15% hoop strain contour were gauged. Based on previous leaker history at Turkey Point Unit 4 and at similar units, as well as previous gauging results, the gauging program also included wedge and patch plate regions. Additionally, when a restricted tube was found close to the inspection boundary, the inspection was expanded in that area. Gauging was also performed on cold leg tubes in all three steam generators in conjunction with the U-bend inspection program conducted from the cold leg side. The inspection for wastage was performed in accordance with the provisions of Regulatory Guide 1.83.

Handhole inspections of the visible tube support plates using photographs were performed in all three steam generators in order to assess the support plate conditions.

Results of Inspection and Corrective Action

No leaking tubes were observed in any steam generator during this inspection. Also, no tube leaks have occurred over the last six months of operation.

Gauging results indicated that any tube near the tube lane which restricted the 0.650" probe was within the 15% hoop strain contour. In addition, tubes restricting the 0.540" probe were within the 17.5% hoop strain contour boundary. In the tubelane region there were two tubes in the three steam generators that restricted the 0.540" eddy current probe. Activity was noted in wedge areas including the cold leg wedge areas inspected. This was the first time wedge regions on the cold leg side which were inspected. It appears that the growth of magnetite and associated denting are following the similar patterns in the hot leg wedge regions. The growth of magnetite and tube denting on the hot leg side are consistent with experience at other units. Indicated tube restrictions on the cold leg side in the tubelane region fell within appropriate strain contour boundaries and were adjacent to previous denting. The implementation of the plugging criteria discussed below combined with previous plugging for various causes, resulted in a total of approximately 18.7% of the tubes being plugged.

The Regulatory Guide 1.83 inspection determined that a total of 20 tubes had to be plugged due to wall thinning.

Plugging Criteria

The plugging criteria implemented by the licensee is essentially the same as that used at other units with similarly degraded steam generator conditions. As in the previously accepted plugging criteria; e.g., as those discussed in the SER attached to the Order dated March 8, 1978 (19), FPL has performed preventive plugging based on the projected growth of the critical tube hoop strain contours predicted by the finite element analysis. This same approach has been used to establish the extent of preventive plugging necessary for continued operation of Turkey Point Unit 3 and Surry Units 1 and 2.

The progression of strain contours over the intended operating period is utilized to preventively plug beyond a tube which does not allow passage of a 0.540" probe. The progression of the 17.5% strain contour has been used to define the extent of preventive plugging necessary. This is identical to the criteria applied to Surry Unit 2, following the March 1978 (20), inspection and to Surry Unit 1, following the inspection performed during the April/May, 1978 (21), refueling outage.

SUMMARY

Turkey Point Unit 4 is one of the six lead PWR facilities that were identified to have suffered moderate to extensive tube denting and that have been under close monitoring by the NRC staff following the September 15, 1976 (22) tube failure occurrence at Surry Unit 2. Our Safety Evaluation Report attached to Amendment No. 27 to DPR-31 of Turkey Point Unit 3 dated August 16, 1977 (23), evaluated the background information concerning "denting" of steam generator tubes which has been experienced at Surry Units 1 and 2 and Turkey Point Units 3 and 4. This background is incorporated by reference and remains valid. The information discussed above represents an update on the condition of the steam generators at Turkey Point Unit 4.

The steam generator inspection was performed in accordance with a program that is consistent with previously implemented program at Turkey Point Unit 4 and other units. We consider this inspection is adequate in the establishment of the condition of steam generators at this unit.

The gauging program performed at Turkey Point Unit 4 was essentially the same as the programs performed at Turkey Point Unit 3 and Surry Units 1 and 2. As in the gauging program performed at Surry Unit 2 during March, 1978 (20), and Surry Unit 1 during April/May, 1978 (21), the 15% tube hoop strain contour was used to define the gauging boundary. These gauging programs have been developed over the course of time in consultation with the NRC staff and have been determined to be acceptable. The inspection of the Turkey Point Unit 4 steam generators has demonstrated that the tube degradation which has occurred to date follows the pattern experienced at Turkey Point Unit 3 and Surry Units 1 and 2. Results of this inspection also indicated that not all tubes within the predicted 17.5% strain boundary restricted the 0.540" probe, which demonstrated quantitatively the conservatism in the tube plugging criteria. Furthermore, the results of this inspection at Turkey Point Unit 4 indicates that no unexpected degradation is occurring and that no new phenomena has been uncovered.

The preventive plugging pattern bounds those tubes which may be anticipated to attain the level of strain which could lead to stress corrosion cracking during the next period of operation and provide reasonable assurance that an acceptable margin of safety will be maintained in accordance with Regulatory Guide 1.121. The preventive plugging conducted by the licensee during the current outage justifies operation of the Turkey Point 4 steam generators for an additional six equivalent months.

We have concluded based on the considerations discussed above, that (1) Turkey Point Unit 4 may be operated for an additional six equivalent months under the restrictions delineated in the Amendment to the license to which this SER applies; at the end of this period, Turkey Point Unit 4 is to be shut down, the steam generators are to be reprobed to determine the extent and pattern of additional tube denting and the results of this gauging program are to be submitted for our review and evaluation prior to the resumption of power operation, and (2) because the results of this inspection indicate that no unexpected degradation is occurring, no new phenomenon have been uncovered, the results were within the bounds of previously established criteria and that this change 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; a significant hazards consideration is not involved.

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 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 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.

References

1. Letter from R. E. Uhrig (FPL) to V. Stello (NRC) Serial No. L-78-210, June 19, 1978.

2. Letter from R. E. Uhrig (FPL) to V. Stello (NRC) Serial No. L-78-230, July 10, 1978.

3. Letter from R. E. Uhrig (FPL) to V. Stello (NRC) Serial No. L-78-242, July 20, 1978.

4. Letter from R. E. Uhrig (FPL) to V. Stello (NRC) Serial No. L-78-264, August 9, 1978.

5. Letter from R. E. Uhrig (FPL) to V. Stello (NRC) Serial No. L-78-297, September 13, 1978.

6. R. A. Geroge, et. al., "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974.

7. WCAP-9220, Westinghouse ECCS Evaluation Model February 1978 Version, February 1978.

8. Westinghouse letter NS-CE-1751 (C. Eicheldinger) to NRC (J. F. Stolz), "LOCA ECCS Analysis with Zirc/Water Reaction Corrections," dated April 7, 1978.

9. Westinghouse letter NS-TMA-1830, "Supplementary Information for WCAP-9220," dated June 16, 1978.

10. Westinghouse letter NS-TMA-1834, "Supplementary Information for WCAP-9220," dated June 20, 1978.

11. NRC letter D. F. Ross, Jr. to D. B. Vassallo, "Safety Evaluation Report on Revised Westinghouse ECCS Evaluation Model," dated August 23, 1978.

12. Letter from NRC (A. Schwencer) to Florida Power and Light Company (R. E. Uhrig), dated June 7, 1978 transmitting the Order for Modification of Licenses dated June 7, 1978.

13. Florida Power and Light Company letter L-76-419 (R. E. Uhrig) to NRC (V. Stello), dated December 9, 1976, transmitting Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)."

14. Florida Power and Light Company letter L-77-217 (R. E. Uhrig) to NRC (G. Lear), dated July 11, 1977.

15. Florida Power and Light Company letter L-78-127 (R. E. Uhrig) to NRC (V. Stello), dated April 10, 1978.

16. Letter from R. E. Uhrig (FPL) to V. Stello, NRC, Serial No. 2-78-271, dated August 16, 1978.

References

17. Letter from R. E. Uhrig (FPL) to V. Stello (NRC) Serial No. L-78-91, September 6, 1978.

18. Letter from NRC (A. Schwencer) to FPL (R. E. Uhrig) dated August 3, 1978, transmitting the Order for Modification of License No. DPR-41, dated August 2, 1977 (corrected August 11, 1977).

19. Letter from NRC (A. Schwencer) to FPL (R. E. Uhrig) dated March 8, 1978, transmitting the Order for Modification of License No. DPR-41 dated March 8, 1978.

20. Order for Modification of License dated April 7, 1978 (License DPR-37, Docket No. 50-281).

21. Order for Modifications of License dated June 23, 1978 (License No. DPR-32, Docket No. 50-280).

22. Letter from VEPCO (C. M. Stallings) to NRC (B. C. Rusche) dated October 19, 1976 (Docket No. 50-281).

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 Nos. 38 and 31 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 amendments change the Technical Specifications for Turkey Point Unit Nos. 3 and 4 in connection with the refueling of Unit 4 Cycle 5 operation and authorize the operation of Turkey Point Unit Nos. 3 and 4 with up to an average of 25% of the tubes in the three steam generators in each unit in a plugged condition. In addition, the steam generators in Turkey Point Unit 4 have been inspected by FPL and reported on September 6, 1978. The steam generators have been found satisfactory by the NRC for an additional six equivalent months.

The operating limits regarding the steam generators for Unit 4, which were previously governed by NRC Orders for Modification of License dated August 3 and 11, 1977 and March 8, 1978, are superseded by this amendment.

The requirements of the NRC Order for Modification of Licenses dated June 7, 1978 are satisfied by the licensee's submittal dated August 9,1978 and supplemented on September 13, 1978. The augmented surveillance procedures in the April 10, 1978 letter from FPL and incorporated in the June 7, 1978 order will no longer be required.

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 regultions in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of the steam generator inspection was not required since the amendments do not involve a significant hazards consideration. Notice of Proposed Issuance of Amendments to Facility Operating License in connection with the Unit 4 Cycle 5 reload and the 25% steam generator tube plugging was published in the Federal Register on August 9, 1978 (43 FR 35406). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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 June 19, 1978 , supplemented on July 10 and 20, August 9 and 16 and September 13, 1978; (2) Amendment Nos. 38 and 31 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 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 Operating Reactors.

Dated at Bethesda, Maryland, this 22nd day of September, 1978

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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