

February 20, 1996

DISTRIBUTION
See attached sheet

Mr. J. H. Goldberg
President-Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

**SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS RE:
IMPLEMENTAION OF THE REVISED THERMAL DESIGN PROCEDURE AND STEAM
GENERATOR WATER LEVEL LOW-LOW SETPOINT (TAC NOS. M92208 AND M92209)**

Dear Mr. Goldberg:

The Commission has issued the enclosed Amendment No. 183 to Facility Operating License No. DPR-31 and Amendment No. 177 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 (TS) in response to your application dated May 5, 1995, as supplemented by letter dated September 28, 1995, relating to implementaion of a revised thermal design procedure and steam generator water level low-low setpoint.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Richard P. Croteau, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-250
and 50-251

Enclosures:

1. Amendment No. 183 to DPR-31
2. Amendment No. 177 to DPR-41
3. Safety Evaluation

cc w/enclosures: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

February 20, 1996

Mr. J. H. Goldberg
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS RE:
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GENERATOR WATER LEVEL LOW-LOW SETPOINT (TAC NOS. M92208 AND M92209)

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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

A handwritten signature in black ink, appearing to read "R. Croteau".

Richard P. Croteau, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-250
and 50-251

Enclosures:

1. Amendment No. 183 to DPR-31
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3. Safety Evaluation

cc w/enclosures:
See next page

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DATED: February 20, 1996

AMENDMENT NO. 183 TO FACILITY OPERATING LICENSE NO. DPR-31-TURKEY POINT UNIT 3
AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE NO. DPR-41-TURKEY POINT UNIT 4

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Docket File

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
License No. DPR-31

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Florida Power and Light Company (the licensee) dated May 5, 1995, as supplemented by letter dated September 28, 1995 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 and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 183, are hereby incorporated in the license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION



David B. Matthews, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 20, 1996



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER AND LIGHT COMPANY
DOCKET NO. 50-251
TURKEY POINT PLANT UNIT NO. 4
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 177
License No. DPR-41

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Florida Power and Light Company (the licensee) dated May 5, 1995, as supplemented by letter dated September 28, 1995, 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 and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 177, are hereby incorporated in the license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION



David B. Matthews, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 20, 1996

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 183 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 177 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove pages

Insert pages

2-2

2-2

2-4

2-4

2-5

2-5

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3/4 3-27

B 3/4 2-8

B 3/4 2-8

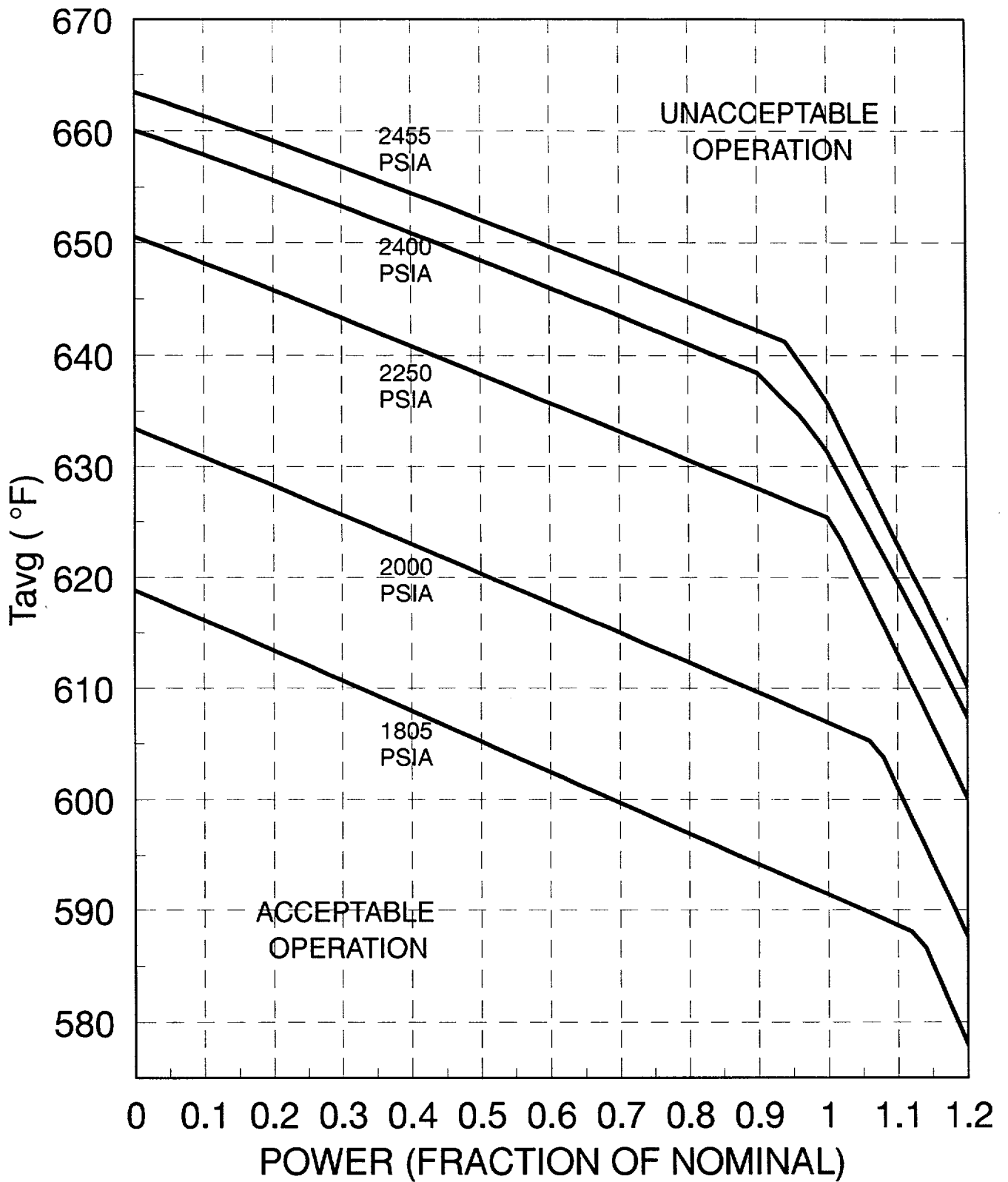


Figure 2.1-1
Reactor Core Safety Limit - Three Loops in
Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
1. Manual Reactor Trip	N.A	N.A.
2. Power Range, Neutron Flux a. High Setpoint b. Low Setpoint	≤112.0% of RTP** ≤28.0% of RTP**	≤109% of RTP** ≤25% of RTP**
3. Intermediate Range, Neutron Flux	≤31.0% of RTP**	≤25% of RTP**
4. Source Range, Neutron Flux	≤1.4 X 10 ⁵ cps	≤10 ⁵ cps
5. Overtemperature ΔT	See Note 2	See Note 1
6. Overpower ΔT	See Note 4	See Note 3
7. Pressurizer Pressure-Low	≥1817 psig	≥1835 psig
8. Pressurizer Pressure-High	≤2403 psig	≤2385 psig
9. Pressurizer Water Level-High	≤92.2% of instrument span	≤92% of instrument span
10. Reactor Coolant Flow-Low	≥88.8% of loop design flow*	≥90% of loop design flow*
11. Steam Generator Water Level Low-Low	≥8.9% of narrow range instrument span	≥10% of narrow range instrument span

* Loop design flow = 89,500 gpm

** RTP = Rated Thermal Power

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
12. Steam/Feedwater Flow Mismatch Coincident With Steam Generator Water Level-Low	Feed Flow $\leq 23.9\%$ below rated Steam Flow $\geq 8.9\%$ of narrow range instrument span	Feed Flow $\leq 20\%$ below rated Steam Flow $\geq 10\%$ of narrow range instrument span
13. Undervoltage - 4.16 kV Busses A and B	$\geq 69\%$ bus voltage	$\geq 70\%$ bus voltage
14. Underfrequency - Trip of Reactor Coolant Pump Breaker(s) Open	≥ 55.9 Hz	≥ 56.1 Hz
15. Turbine Trip		
a. Auto Stop Oil Pressure	≥ 42 psig	≥ 45 psig
b. Turbine Stop Valve Closure	Fully Closed***	Fully Closed***
16. Safety Injection Input from ESF	N. A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 6.0 \times 10^{-11}$ amps	Nominal 1×10^{-10} amp

*** Limit switch is set when Turbine Stop Valves are fully closed.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left\{ \frac{1 + \tau_1 S}{1 + \tau_2 S} \right\} \left(\frac{1}{1 + \tau_3 S} \right) \leq \Delta T_o \left\{ K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S} \right) - T' \right] + K_3 (P - P') - f_1(\Delta T) \right\}$$

Where: ΔT = Measured ΔT by RTD Instrumentation

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead/Lag compensator on measured ΔT ; $\tau_1 = 0s$, $\tau_2 = 0s$

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT ; $\tau_3 = 0s$

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 = 1.25;

K_2 = 0.016/ $^{\circ}F$;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg} , $\tau_4 = 25s$, $\tau_5 = 3 s$;

T = Average temperature, $^{\circ}F$;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg} ; $\tau_6 = 0s$

T' \leq 574.2 $^{\circ}F$ (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.0011/psig;

P = Pressurizer pressure, psig;

TABLE 2.2-1 (Continued)TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

- P' \geq 2235 psig (Nominal RCS operating pressure);
- S = Laplace transform operator, s^{-1} ;

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

- (1) For $q_t - q_b$ between - 46% and + 2%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;
- (2) For each percent that the magnitude of $q_t - q_b$ exceeds - 46%, the ΔT Trip Setpoint shall be automatically reduced by 1.5% of its value at RATED THERMAL POWER; and
- (3) For each percent that the magnitude of $q_t - q_b$ exceeds + 2%, the ΔT Trip Setpoint shall be automatically reduced by 2.3% of its value at RATED THERMAL POWER.

NOTE 2: The channels maximum trip setpoint shall not exceed its computed setpoint by more than 0.73% of instrument span.

TABLE 2.2-1 (Continued)TABLE NOTATIONS (Continued)NOTE 3: OVERPOWER ΔT

$$\Delta T \left\{ \frac{1 + \tau_1 S}{1 + \tau_2 S} \right\} \left(\frac{1}{1 + \tau_3 S} \right) \leq \Delta T_0 \left\{ K_4 - K_5 \frac{(\tau_7 S)}{1 + \tau_7 S} \left(\frac{1}{1 + \tau_6 S} \right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S} \right) - T'' \right] - f_2(\Delta T) \right\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 \leq 1.10,

K_5 \geq 0.02/ $^{\circ}$ F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constants utilized in the lead-lag compensator for T_{avg} , $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

TABLE 2.2-1 (Continued)TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	$0.00232/^{\circ}\text{F}$ for $T > T''$
	=	0 for $T \leq T''$,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq 574.2^{\circ}\text{F}$),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than 0.4% of instrument span.

POWER DISTRIBUTION LIMIT

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

- a. Reactor Coolant System T_{avg} $\leq 576.6^{\circ}\text{F}$
- b. Pressurizer Pressure ≥ 2209 psig*, and
- c. Reactor Coolant System Flow $\geq 277,900$ gpm

APPLICABILITY: MODE 1.

ACTION:

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 within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 RCS flow rate shall be monitored for degradation at least once per 12 hours.

4.2.5.3 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.4 After each fuel loading, and at least once per 18 months, the RCS flow rate shall be determined by precision heat balance after exceeding 90% RATED THERMAL POWER. The measurement instrumentation shall be calibrated within 90 days prior to the performance of the calorimetric flow measurement. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

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

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
5. Feedwater Isolation (Continued)		
c. Steam Generator Water Level High-High	≤81.9% of narrow range instrument span	≤80% of narrow range instrument span
6. Auxiliary Feedwater (3)		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	≥8.9 of narrow range instrument span.	≥10% of narrow range instrument span.
c. Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
d. Bus Stripping	See Item 7. below for all Bus Stripping Allowable Values.	See Item 7. below for all Bus Stripping Trip Setpoints.
e. Trip of All Main Feedwater Pump Breakers	N.A.	N.A.
7. Loss of Power		
a. 4.16 kV Busses A and B (Loss of Voltage)	N.A.	N.A.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q(Z)$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or incore thermocouple map are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR above the applicable design limits throughout each analyzed transient. The indicated T_{avg} value of 576.6°F and the indicated pressurizer pressure value of 2209 psig correspond to analytical limits of 578.2°F and 2185 psig respectively, with allowance for measurement uncertainty.

The indicated RCS flow value of 277,900 gpm corresponds to an analytical limit of 268,500 gpm which is assumed to have a 3.5% measurement uncertainty. The above measurement uncertainty estimates assume that these instrument channel outputs are averaged to minimize the uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to ensure that the DNB-related flow assumption is met and to ensure correlation of the flow indication channels with measured flow. Six month drift effects have been included for feedwater temperature, feedwater flow, steam pressure, and the pressurizer pressure inputs. The flow measurement is performed within ninety days of completing the cross-calibration of the hot leg and cold leg narrow range RTDs. The indicated percent flow surveillance on a 12-hour basis will provide sufficient verification that flow degradation has not occurred. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 183 TO FACILITY OPERATING LICENSE NO. DPR-31
AND AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By letter dated May 5, 1995, as supplemented by letter dated September 28, 1995, Florida Power and Light (FPL or the licensee) proposed a change to the Technical Specifications (TS) for Turkey Point Units 3 and 4. The submittal describes the TS changes (TS 2.2.1, 2.2, 3/4.2.5, 3/4.3.2, and associated bases) proposed by FPL. The proposed revision to the TS includes (a) the implementation of Westinghouse's NRC-approved Revised Thermal Design Procedure (RTDP), and (b) a revision to the steam generator water level low-low trip setpoint. The technical justifications for these revisions are given below.

The traditional method for accounting for the design modeling uncertainties that enter into the determination of a departure from nucleate boiling ratio (DNBR) assumes that key input parameters of the core thermal-hydraulic code are simultaneously at their worst level of uncertainty (a "deterministic method"). The RTDP was developed to remove the excess conservatism of the deterministic method in that variations in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and DNB correlation predictions are considered statistically to obtain a departure from nucleate boiling (DNB) uncertainty factor. The methodology, therefore, assumes that the uncertainty associated with the DNB correlation can be statistically combined with the plant parameters uncertainties. The net result is a large increase in DNB margin (i.e., the difference between the design limit and the safety analysis limit DNBR).

The RTDP is a statistical procedure that combines the uncertainties of power, pressure, temperature, and flow with the DNBR correlation uncertainty to calculate the DNBR design limit. This DNBR calculation is based on new input parameters for conservative plant operation such as reduced reactor coolant system (RCS) flow and an increase in F^{AH} . These calculations led to new core thermal limits and overtemperature delta T and overpower delta T reactor trip set points. The methodology used in this license amendment is the same as that approved for use by South Carolina Electric & Gas Company, licensee for the Virgil C. Summer Nuclear Station.

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In addition, the licensee proposed to relax the DNBR monitoring Technical Specification to accurately reflect the intent of the specification and the capability of monitoring DNB parameters.

The licensee also proposed to revise the steam generator water level low-low setpoints based on the inclusion of additional process measurement accuracy terms and NRC-accepted methodology.

2.0 TECHNICAL EVALUATION

2.1 Revised Thermal Design Procedure

The licensee requested the TS changes regarding the utilization of the RTDP to (1) generate revisions to the core safety limits, namely, TS Figure 2.1-1, and overtemperature and overpower delta T setpoints and associated uncertainties. These revisions will provide an additional operating margin for the core safety limits and are reflected in TS Tables 2.2-1 and 3.3-3 and (2) to determine additional operation margin in the DNB parameters and the loss of flow setpoint identified in TS 3/4.2.5 and 2.2.1.

By making changes to surveillance procedures, implementing upgraded process instrumentation equipment, and using the RDTP methodology, the licensee has been able to increase the operational margin to the limits associated with the measurement and indication of RCS flow.

Revision of the thermal limits is reflected in Figure 2.1-1 of the submittal. These thermal limits were generated by using the RTDP Methodology. The RTDP takes advantage of the conservative use of the statistical combination of values for the reactor power, RCS flow, and temperature and pressure to calculate the DNBR limit. Changes in the hot channel factors and the RCS flow cause the DNB core limits to change. The RTDP methodology is a modification of the methodology defined in WCAP-8567 ("Improved Thermal Design Procedure") and approved by the NRC.

Subsequent to changes of the core operating limits, the overtemperature and overpressure delta T reactor trip setpoints, and their associated uncertainties were calculated on the basis of the new core thermal limits. The licensee conducted a review of the Turkey Point updated final safety analysis report (UFSAR) to determine those events sensitive to changes in overtemperature and overpower delta T reactor trip points. Each of the events (i.e., rod withdrawal at power, boron dilution, and loss of load) were analyzed by the licensee to determine if the various acceptance criteria were met. In all cases, the acceptance criteria were met, and therefore, the margin of safety was maintained. Since NRC-accepted methodology was used and the various acceptance criteria were met, the staff finds the proposed changes acceptable.

The values of the setpoints used in the UFSAR as well as the reactor core thermal limits revision, are reflected in the revised TS submitted by the licensee.

The overtemperature and overpower delta T trip function values in TS Table 2.1-1 were revised to reflect the utilization of the RTDP methodology. The overtemperature delta T trip function provides core protection from DNB for all combinations of pressure, power, flow, and coolant temperature when the coolant pressure is within the range defined by the pressurizer's high-and-low pressure trips.

The overpower delta T reactor trip function is also defined in Table 2.2-1 of the submittal. This function is designed specifically to ensure operation within the fuel centerline temperature design limit. Analyses conducted by the licensee showed that indeed this is the case with gross core thermal power controlled within a prescribed limit (118 percent of nominal full power).

The justification for the above TS changes is the use of the RTDP Methodology which provides additional margin to Turkey Point's TS by generating increased DNB margin. The existing setpoints are relaxed to provide operational benefits using additional margin gained by implementing the RTDP. The RTDP Methodology takes advantage of the conservative use of statistical combination of values for reactor power, RCS flow, temperature and pressure to calculate the DNBR limit. This methodology was approved by the NRC in WCAP-11397-P-A. The values of the setpoints used in the safety analysis as well as the Reactor Core Limits revision are reflected in the revised TS. The revised Note 2 changed the channels maximum trip setpoint to reflect changes to the Allowable Value as calculated utilizing the methodology defined in Westinghouse Topical Report WCAP-12745 Setpoint Methodology.

The proposed changes to the overtemperature and overpower delta T reactor trip functions do not reduce the margin of safety because the margin of safety associated with these functions, as verified by the accident analyses, was found to be well within the acceptable limits. The licensee reanalyzed all the transients affected by the implementation of the RTDP methodology and found that they met the applicable analyses acceptance criteria.

Since NRC-accepted methodology was used and the various acceptance criteria were met, the staff finds the proposed changes acceptable.

2.2 Steam Generator Process Measurement Accuracy (PMA)

Westinghouse has identified that a potential exists that sufficient margin may not have been included in the PMA term for the Steam Generator Water Level (SGWL) instrumentation uncertainty calculations. This insufficient margin would impact the protection functions which use this parameter, i.e., Reactor Protection System-Steam Generator Water Level Low, Reactor Trip and Engineered Safety Feature Actuation System-Steam Generator Water Level Low-Low, Actuation of the Auxiliary Feedwater System. Therefore, the PMA uncertainty terms and corresponding protection system setpoints have been recalculated to account for additional uncertainties.

Previously a random value of + or - 2.0% span was used for the PMA term in setpoint uncertainty calculations for all models of steam generator design. This value was based on various process parameters which are determined by specific plant operating conditions. In June 1991, an Instrument Society of America (ISA) paper identified additional errors that should be considered. These errors are associated with: (1) reference leg temperature changes from calibration temperature, (2) downcomer subcooling, and (3) fluid velocity effects. These errors are not considered random in nature, and therefore, are treated as biases. With the inclusion of additional PMA terms and their treatment as a bias, the existing trip setpoints for SGWL Low and Low-Low provide margin to protect the existing safety analysis limits.

The following values are revised for TS Table 2.2-1 "Reactor Trip System Instrumentation Trip setpoints," Functional Unit 11, Steam Generator Water Level Low-Low, and Functional Unit 12, Steam/Feedwater Flow Mismatch Coincident With Steam Generator Water Level-Low:

- a) Trip Setpoint to read " $\geq 10\%$ " instead of " $\geq 15\%$ "
- b) Allowable Value to read " $\geq 8.9\%$ " instead of " $\geq 13.3\%$ "

The following values are revised for Table 3.3.3 "Engineered Safety Features Actuation System Instrumentation Trip Setpoints," Functional Unit 6, Steam Generator Water Level Low-Low:

- a) Trip Setpoint to read " $\geq 10\%$ " instead of " $\geq 15\%$ "
- b) Allowable Value to read " 8.9%" instead of " 13.3%," and the sign to read " \geq " instead of " \leq ."

The correct sign for the Functional 6b, Allowable Value, was in the NRC issued license amendments 140/135 on April 23, 1991, (Ref. 8); however, this sign was incorrectly transposed when the NRC issued license amendment 146/141 by letter dated August 26, 1991.

The justification for the above TS changes is the Turkey Point SGWL Low-Low protection setpoints have been recalculated to account for additional PMA uncertainties. In previous calculations the PMA was assumed to be bounded by a + or - 2.0% span allowance. New calculations have been performed which explicitly identified the impact of the PMA term on the channel Statistical Allowance. The reduction in SGWL Low-low trip setpoint has been analyzed in Chapter 14 analyses of the Updated Final Safety Analysis Report (UFSAR) and FPL has confirmed that sufficient margin exists to maintain safe plant operation. In the updated analyses, the SGWL Low-Low setpoint is assumed at 5 percent narrow range span level. These calculations allowed margins to be re-allocated to improve plant operations by reducing the SGWL Low-Low trip setpoint and accompanying Allowable Value while still maintaining a margin of safety.

The staff finds the proposed change acceptable since NRC-approved methodology was used to determine the setpoints and sufficient margin exists to maintain safe plant operation with the proposed revisions.

2.3 Upgraded Process Instrumentation Equipment

The licensee previously replaced portions of the Hagan analog protection and control process instrumentation system with a Westinghouse Electric Corporation Eagle-21 microprocessor based Class 1E protection system. In 1991, the Eagle-21 equipment was installed at Turkey Point Unit 3 and 4 during the modification to remove the resistance temperature detector (RTD) in the RCS Bypass system. The Eagle-21 system is used for Overpower and Overtemperature-Delta-T trips and Pressurizer Level Trips.

During this current safety evaluation, the staff has reviewed the design and application of the Eagle-21 system from July 31, 1988 through July 1995 to ascertain hardware or software problems associated with the Eagle-21 system. Based on this review, the staff concludes that the software and hardware problems have been resolved.

3.0 SUMMARY

The staff has reviewed FPL's submittal pertaining to the analysis of FPL's core thermal limits utilizing the RTDP methodology. These new limits are reflected in Figure 2.1-1 of their submittal. All the reactor trip setpoints pertaining to overtemperature and overpower delta T were calculated using the RTDP methodology. The staff has previously found use of this methodology acceptable. All the accidents analyzed (rod withdrawal, boron dilution, etc.) were found to meet the acceptance criteria, and they consequently maintain the same margin of safety as before. Therefore, the staff finds the proposed TS changes acceptable.

With respect to the PMA term and corresponding protection system setpoint changes, the staff finds the proposed changes acceptable since sufficient margin exists to maintain safe plant operation.

4.0 STATE CONSULTATION

Based upon the written notice of the proposed amendments, the Florida State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 54719). Accordingly, these

amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSIONS

The staff concluded, based on the considerations discussed in this evaluation, 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 Commission's regulations and (3) issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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