August 24, 1995

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: COLUMN FORMAT FOR REACTOR PROTECTION SYSTEM (RPS) AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) SETPOINTS (TAC NOS. M92402 AND M92403)

Dear Mr. Goldberg:

The Commission has issued the enclosed Amendment No. 176 to Facility Operating License No. DPR-31 and Amendment No. 170 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 23, 1995, relating to revising the column format for the RPS and ESFAS setpoints. Please note that we clarified portions of the TS Bases section following discussions with your staff.

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. 176 to DPR-31
2. Amendment No. 170 to DPR-41
3. Safety Evaluation

cc w/enclosures: See next page

Document Name: G:TURKEY\TP92402.AMD * see previous concurrence

OFFICE	LA:PDII-1	PM:PDII-1	D:PDII-1	HICB	OGC	OGC
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DATE	7/6/95	7/6/95	8/24/95	8/16/95	~~9/13~~/95	8/15/95
COPY	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	

Replace footnote on p. 2 of SE

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

August 24, 1995

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: COLUMN FORMAT FOR REACTOR PROTECTION SYSTEM (RPS) AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) SETPOINTS (TAC NOS. M92402 AND M92403)

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cc w/enclosures:
See next page

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DATED: August 24, 1995

AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. DPR-31-TURKEY POINT UNIT 3
AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-41-TURKEY POINT UNIT 4

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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. 176
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 23, 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-31 is hereby amended to read as follows:

   (B) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 176 , 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David C. Trimble for

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: August 24, 1995

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. 170
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 23, 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. 170, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David C. Trimble, Jr. for

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: August 24, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 176 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 170 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove pages	Insert pages
2-3	2-3
2-4	2-4
2-5	2-5
2-6	2-6
B 2-3	B 2-3
B 2-3a	none
3/4 3-13	3/4 3-13
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
B 3/4 3-1	B 3/4 3-1
B 3/4 3-1a	B 3/4 3-1a

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

Action:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the setpoint consistent with the Trip setpoint value within permissible calibration tolerance.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that the affected channel is OPERABLE; or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip	N.A	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	$\leq$112.0% of RTP**	$\leq$109% of RTP**
b. Low Setpoint	$\leq$28.0% of RTP**	$\leq$25% of RTP**
3. Intermediate Range, Neutron Flux	$\leq$31.0% of RTP**	$\leq$25% of RTP**
4. Source Range, Neutron Flux	$\leq 1.4 \times 10^5$ cps	$\leq 10^5$ cps
5. Overtemperature ΔT	See Note 2	See Note 1
6. Overpower ΔT	See Note 4	See Note 3
7. Pressurizer Pressure-Low	$\geq$1817 psig	$\geq$1835 psig
8. Pressurizer Pressure-High	$\leq$2403 psig	$\leq$2385 psig
9. Pressurizer Water Level-High	$\leq$92.2% of instrument span	$\leq$92% of instrument span
10. Reactor Coolant Flow-Low	$\geq$88.7% of loop design flow*	$\geq$90% of loop design flow*
11. Steam Generator Water Level Low-Low	$\geq$13.2% of narrow range instrument span	$\geq$15% 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

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
12. Steam/Feedwater Flow Mismatch Coincident With	Feed Flow $\leq$23.9% below rated Steam Flow	Feed Flow $\leq$20% below rated Steam Flow
Steam Generator Water Level-Low	$\geq$13.2% of narrow range instrument span	$\geq$15% 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	$\geq$55.9 Hz	$\geq$56.1 Hz
15. Turbine Trip		
a. Auto Stop Oil Pressure	$\geq$42 psig	$\geq$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	$\geq 6.0\times10^{-11}$ amps	Nominal 1×10^{-10} amp

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	$\leq$13.0% RTP**	Nominal 10% of RTP**
2) Turbine First Stage Pressure	$\leq$13.0% Turbine Power	Nominal 10% Turbine Power
c. Power Range Neutron Flux, P-8	$\leq$48.0% RTP**	Nominal 45% of RTP**
d. Power Range Neutron Flux, P-10	$\geq$7.0% RTP**	Nominal 10% of RTP**
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.

** RTP = RATED THERMAL POWER

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the Nominal Trip Setpoint when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift that may occur between operational tests and the accuracy to which setpoints can be measured and calibrated, statistical allowances are provided for in the Nominal Trip Setpoint and Allowable Values in accordance with the setpoint methodology described in WCAP's 12201 and 12745. Surveillance criteria have been determined and are controlled in Plant procedures and in design documents. The surveillance criteria ensure that instruments which are not operating within the assumptions of the setpoint calculations are identified. An instrument channel is considered OPERABLE when the surveillance is within the Allowable Value and the channel is capable of being calibrated in accordance with Plant procedures. Sensor and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

The inability to demonstrate through measurement and/or analytical means, using the methods described in WCAP's 12201 and 12745 ($TA \geq R+S+Z$), that the Reactor Trip function would have occurred within the values specified in the design documentation provides a threshold value for REPORTABLE EVENTS.

There is a small statistical probability that a properly functioning device will drift beyond determined surveillance criteria. Infrequent drift outside the surveillance criteria are expected. Excessive rack or sensor drift that is more than occasional, may be indicative of more serious problems and should warrant further investigations.

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value within permissible calibration tolerance.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-3, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-3 and determine within 12 hours that the affected channel is OPERABLE; or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection (Reactor Trip, Turbine Trip, Feedwater Isolation, Control Room Ventilation Isolation, Start Diesel Generators, Containment Phase A Isolation (except Manual SI), Containment Cooling Fans, Containment Filter Fans, Start Sequencer, Component Cooling Water, Start Auxiliary Feedwater and Intake Cooling Water)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.
c. Containment Pressure--High	≤4.5 psig	≤4.0 psig
d. Pressurizer Pressure--Low	≥1712 psig	≥1730 psig
e. High Differential Pressure Between the Steam Line Header and any Steam Line.	≤114 psig	≤100 psi
f. Steam Line Flow--High	≤A function defined as follows: A ΔP corresponding to 42.6% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 122.6% steam flow at full load.	≤A function defined as follows: A ΔP corresponding to 40% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 120 % steam flow at full load.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
Coincident with: Steam Generator Pressure--Low	≥588 psig	≥614 psig
or T_{avg}--Low	≥542.5°F	≥543°F
2. Containment Spray		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Containment Pressure--High-High Coincident with:	≤22.6 psig	≤20.0 psig
Containment Pressure--High	≤4.5 psig	≤4.0 psig
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1 above for all Safety Injection Allowable Values.	See Item 1 above for all Safety Injection Trip Setpoints.
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
3. Containment Isolation (Continued)		
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High Coincident with:	$\leq$22.6 psig	$\leq$20.0 psig
Containment Pressure--High	$\leq$4.5 psig	$\leq$4.0 psig
c. Containment Ventilation Isolation		
1) Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
4) Containment Radio-activity--High (1)	Particulate (R-11) $\leq 6.8 \times 10^5$ CPM Gaseous (R-12) See Note 2	Particulate (R-11) $\leq 6.1 \times 10^5$ CPM Gaseous (R-12) See Note 2
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation (Continued)		
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High Coincident with:	$\leq$22.6 psig	$\leq$20.0 psig
Containment Pressure--High	$\leq$4.5 psig	$\leq$4.0 psig
d. Steam Line Flow--High	$\leq$A function defined as follows: A ΔP corres-ponding to 42.6% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 122.6% steam flow at full load.	$\leq$A function defined follows: A ΔP corres-ponding to 40% steam flow at 0% load in-creasing linearly from 20% load to a value corresponding to 120% steam flow at full load.
Coincident with: Steam Line Pressure--Low	$\geq$588 psig	$\geq$614 psig
or		
T_{avg}--Low	$\geq$542.5°F	$\geq$543°F
5. Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
5. Feedwater Isolation (Continued)		
c. Steam Generator Water Level High-High	$\leq$81.9% of narrow range instrument span	$\leq$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	$\leq$13% of narrow range instrument span.	$\geq$15% 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.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE #	TRIP SETPOINT
7. Loss of Power (Continued)		
b. 480V Load Centers Degraded Voltage		
Load Center		
3A	[ ]	430V ±5V (10 sec ±1 sec delay)
3B	[ ]	438V ±5V (10 sec ±1 sec delay)
3C	[ ]	434V ±5V (10 sec ±1 sec delay)
3D	[ ]	434V ±5V (10 sec ±1 sec delay)
4A	[ ]	435V ±5V (10 sec ±1 sec delay)
4B	[ ]	434V ±5V (10 sec ±1 sec delay)
4C	[ ]	434V ±5V (10 sec ±1 sec delay)
4D	[ ]	430V ±5V (10 sec ±1 sec delay)
Coincident with: Safety Injection and	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
Diesel Generator Breaker Open	N.A.	N.A.

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE #	TRIP SETPOINT
7. Loss of Power (Continued)		
c. 480V Load Centers Degraded Voltage		
Load Center		
3A	[ ]	424V ±5V (60 sec ±30 sec delay)
3B	[ ]	427V ±5V (60 sec ±30 sec delay)
3C	[ ]	437V ±5V (60 sec ±30 sec delay)
3D	[ ]	435V ±5V (60 sec ±30 sec delay)
4A	[ ]	430V ±5V (60 sec ±30 sec delay)
4B	[ ]	436V ±5V (60 sec ±30 sec delay)
4C	[ ]	434V ±5V (60 sec ±30 sec delay)
4D	[ ]	434V ±5V (60 sec ±30 sec delay)
Coincident with: Diesel Generator Breaker Open	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
8. Engineering Safety Features Actuation System Interlocks		
a. Pressurizer Pressure	$\leq$2018 psig	Nominal 2000 psig
b. Tavg--Low	$\geq$542.5°F	Nominal 543°F
9. Control Room Ventilation Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
c. Containment Radioactivity--High (1)	Particulate (R-11) $\leq 6.8 \times 10^5$ CPM Gaseous (R-12) See Note 2	Particulate (R-11) $\leq 6.1 \times 10^5$ CPM Gaseous (R-12) See Note 2
d. Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.
e. Air Intake Radiation Level	$\leq$2.83 mR/hr	$\leq$2 mR/hr

TABLE 3.3-3 (continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

(1) Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

(2) Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(F)}$ CPM,

Containment Gaseous Monitor Allowable Value = $\frac{(3.5 \times 10^4)}{(F)}$ CPM,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in Specification 3.11.2.1.

(3) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

If no Allowable Value is specified, as indicated by [], the trip setpoint shall also be the allowable value.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance (due to plant specific design, pulling fuses and using jumpers may be used to place channels in trip), and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Under some pressure and temperature conditions, certain surveillances for Safety Injection cannot be performed because of the system design. Allowance to change modes is provided under these conditions as long as the surveillances are completed within specified time requirements.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-3 are the nominal values at which the bistables are set for each functional unit. The setpoint is considered to be adjusted consistent with the Nominal Trip Setpoint when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift that may occur between operational tests and the accuracy to which setpoints can be measured and calibrated, statistical allowances are provided for in the Nominal Trip Setpoint and Allowable Values in accordance with the setpoint methodology described in WCAP's 12201 and 12745. Surveillance criteria have been determined and are controlled in Plant procedures and in design documents. The surveillance criteria ensure that instruments which are not operating within the assumptions of the setpoint calculations are identified. An instrument channel is considered OPERABLE when the surveillance is within the Allowable Value and the channel is capable of being calibrated in accordance with Plant procedures. Sensor and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

The inability to demonstrate through measurement and/or analytical means, using the methods described in WCAP's 12201 and 12745 ($TA \geq R+S+Z$), that the Reactor Trip function would have occurred within the values specified in the design documentation provides a threshold value for REPORTABLE EVENTS.

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

There is a small statistical probability that a properly functioning device will drift beyond determined surveillance criteria. Infrequent drift outside the surveillance criteria are expected. Excessive rack or sensor drift that is more than occasional, may be indicative of more serious problems and should warrant further investigations.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents events, and transients. Once the required logic combination is completed, the system sends actuation signals to

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO. 170 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 23, 1995, Florida Power and Light Company (FPL or the licensee) proposed a change to the Technical Specifications (TS) for Turkey Point Units 3 and 4. The changes requested involved modifying TS 2.2, "Limiting Safety System Settings, Reactor Protection System Instrumentation Setpoints" and Technical Specification 3/4.3.2, "Engineered Safety Features Actuation System Instrumentation Limiting Condition for Operation" and their associated bases. The proposed change would use a revised setpoint presentation format for the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) instrumentation setpoints contained in Technical Specification Tables 2.2-1 and 3.3-3 while retaining the approved Westinghouse five-column instrument setpoint methodology currently being used for establishing those setpoints. The licensee stated that the intent of the amendments is to eliminate the need for minor administrative license amendments to these tables that do not impact either the Trip Setpoints or the Safety Analysis Limits. Controlled plant drawings contain, and will continue to contain, the necessary information concerning the calculation of each setpoint using the previously approved setpoint methodology. Changes to the "Z", "S" and "TA" terms contained in the controlled plant drawings can only be changed under the formal controls of TS 6.8. The change would result in a relocation of this information from the TS to controlled plant drawings and procedures; however, the information already exists in these documents.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 Fed. Reg. 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies §182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TS, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; and (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.[1] As a result, existing TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

3.0 EVALUATION

The proposed amendment is intended to limit the TS changes required due to changes in plant instrumentation. The values for S, Z, and TA will be removed from the TS along with Equation 2.2-1, ($Z+R+S\leq TA$). The values for S, Z, and TA will be controlled administratively by the licensee through drawings and procedures. The licensee intends to maintain the Westinghouse instrument setpoint methodology that was previously used with the 5 column format. Equation $Z+R+S\leq TA$ will continue to provide the licensee with an optional method

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[1]The Commission recently adopted amendments to 10 CFR 50.36, pursuant to which the rule was revised to codify and incorporate these criteria. See Final Rule, "Technical Specifications," 60 Fed. Reg. 36953 (July 19, 1995). The Commission indicated that reactor core isolation cooling, isolation condenser, residual heat removal, standby liquid control, and recirculation pump are to be included in the TS under Criterion 4, although it recognized that other structures, systems and components could also meet this criterion. 60 Fed. Reg. at 36956.

for operability determination. The bases of the TS will be revised to reflect the proposed changes, but will continue to reference the Westinghouse 5 column setpoint methodology (WCAP-12725 and 12201) including equation $Z+R+S \leq TA$. No changes to setpoints are included in the licensee amendment. Based on the fact that the licensee will continue to use the approved Westinghouse methodology, the staff finds the proposed licensee amendment acceptable. The staff notes that the 2 column format is consistent with the format of the improved standard TS. The control of the relocated information is via the requirements of TS 6.8.

TS 6.8.1 requires that written procedures shall be established, implemented, and maintained covering the items recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 (RG 1.33). RG 1.33, Appendix A, specifies procedures for surveillance tests and calibrations for the Emergency Core Cooling and Reactor Protection Systems. TS 6.8.2 requires that changes to these procedures shall be reviewed and approved prior to implementation in accordance with a formal program described in TS 6.5.3, including a determination of whether or not an unreviewed safety question is involved. The staff finds these controls on the relocated items to be adequate. In addition, the staff notes that the licensee stated in their submittal that any changes to the "Z", "S", and "TA" terms would only be made following a 10 CFR 50.59 evaluation to determine if prior NRC approval is necessary.

The TS Bases were modified from those submitted on May 23, 1995, in order to clarify that the existing Westinghouse methodology would continue to be used by the licensee. This commitment was made in the submittal, but not specifically mentioned in the proposed Bases. The licensee agreed to the revised Bases in a conversation on June 28, 1995.

4.0 CONCLUSIONS

The staff reviewed the proposed changes and determined that the removal of these details do not eliminate the requirements for the licensee to ensure that the system, structure, or component is capable of performing its safety function. Although this information is removed from the TS and incorporated into the administratively controlled documents, the licensee must continue to evaluate any plant modifications that affect any of these components in accordance with TS 6.8. NRC inspection and enforcement programs also enable the staff to monitor facility changes and licensee adherence to commitments and to take any remedial action that may be appropriate.

Based on this review, the staff concluded that 10 CFR 50.36 does not require this information to be retained in the TS. Requirements related to operability, applicability, and surveillance requirements, including performance of testing to ensure operability, are retained due to their importance in mitigating the consequences of an accident. However, the staff determined that the inclusion of this information is an operational detail related to the licensee's safety analysis, which are adequately controlled by the requirements of TS 6.8. Therefore, the continued processing of license amendments related to revisions of the affected tables would afford no significant benefit with regard to protecting the public health and safety.

The staff has concluded, therefore, that removal of this information is acceptable because (1) inclusion in the TS is not specifically required by 10 CFR 50.36 or other regulations, (2) the information has been incorporated into the administratively controlled document, and (3) changes are adequately controlled by TS 6.8.

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 STATE CONSULTATION

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

6.0 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. 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 32364). 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.

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