September 25, 1991

Docket Nos. 50-250
and 50-251

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

Dear Mr. Goldberg:

SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS RE: ADMINISTRATIVE CHANGES (TAC NOS. 80526 AND 80526)

The Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-31 and Amendment No. 144 to Facility Operating License No. DPR-41 for the Turkey Point Plant, Units Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated May 28, 1991.

These amendments revise the Technical Specifications by (1) removing outdated material, (2) incorporating administrative clarifications, and (3) correcting typographical errors.

Two of your requested changes have been denied by the staff. The requested changes would have replaced the word "generators" with the word "lines" on pages 3/4 3-15 and 3/4 3-18. The requested changes are not consistent with the interpretation of NUREG-0452, Westinghouse Standard Technical Specifications, and are therefore not acceptable. The Notice of Denial, which has been forwarded to the Office of the Federal Register for publication, is enclosed.

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)

Rajender Auluck, Sr. Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 149 to DPR-31
2. Amendment No. 144 to DPR-41
3. Safety Evaluation
4. Notice of Denial

cc w/enclosures:
See next page

OP/1

LA:PDII-2	INT:PDII-2	PM:PDII-2	D:PDII-2	OGC
DMiller	FTalbot	RAuluck:kdj	HBerkow	MZOBLER
8/29/91	8/29/91	8/30/91	8/?/91	9/3/91

*MEMO TP 80525/26

DF01
1/1

cc:
Harold F. Reis, Esquire
Newman and Holtzinger, P.C.
1615 L Street, N.W.
Washington, DC 20036

Jack Shreve, Public Counsel
Office of the Public Counsel
c/o The Florida Legislature
111 West Madison Avenue, Room 812
Tallahassee, Florida 32399-1400

John T. Butler, Esquire
Steel, Hector and Davis
4000 Southeast Financial Center
Miami, Florida 33131-2398

Mr. Thomas F. Plunkett, Site
Vice President
Turkey Point Nuclear Plant
Florida Power and Light Company
P.O. Box 029100
Miami, Florida 33102

Joaquin Avino
County Manager of Metropolitan
Dade County
111 NW 1st Street, 29th Floor
Miami, Florida 33128

Senior Resident Inspector
Turkey Point Nuclear Generating Station
U.S. Nuclear Regulatory Commission
Post Office Box 1448
Homestead, Florida 33090

Mr. Jacob Daniel Nash
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

Mr. Gordon Guthrie, Director
Emergency Management
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

Administrator
Department of Environmental
Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W. Suite 2900
Atlanta, Georgia 30323

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

Plant Manager
Turkey Point Nuclear Plant
Florida Power and Light Company
P.O. Box 029100
Miami, Florida 33102

Mr. R. E. Grazio
Director, Nuclear Licensing
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

DATED: September 25, 1991

AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-31-TURKEY POINT UNIT 3
AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. DPR-41-TURKEY POINT UNIT 4

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
F. Talbot
R. Auluck (2)
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
G. Hill (8), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
PD Plant-specific file [Gray File]
M. Sinkule, R-II
Others as required

cc: Plant Service list

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
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 28, 1991, 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) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 149 , 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 the date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Rajender Auluck

for Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: September 25, 1991

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 144
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 28, 1991, 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. 144, 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 the date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Rajender Auluck

for Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 25, 1991

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 149 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 144 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-2	1-2
2-10	2-10
3/4 1-23	3/4 1-23
3/4 3-15	3/4 3-15
3/4 3-18	3/4 3-18
3/4 3-26	3/4 3-26
3/4 3-36	3/4 3-36
3/4 3-43	3/4 3-43
3/4 3-48	3/4 3-48
3/4 3-49	3/4 3-49
3/4 3-53	3/4 3-53
3/4 3-56	3/4 3-56
3/4 3-59	3/4 3-59
3/4 3-60	3/4 3-60
3/4 3-61	3/4 3-61
3/4 4-22	3/4 4-22
3/4 4-28	3/4 4-28
3/4 6-10	3/4 6-10
3/4 6-15	3/4 6-15
3/4 7-33	3/4 7-33
3/4 9-12	3/4 9-12
3/4 9-13	3/4 9-13
3/4 9-15	3/4 9-15
3/4 11-15	3/4 11-15
3/4 12-2	3/4 12-2
B 3/4 9-4	B 3/4 9-4
5-5	5-5
5-6	5-6
6-1	6-1
6-7	6-7
6-14	6-14
6-15	6-15
6-17	6-17
6-22	6-22

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specification 3.6.4.

b. The equipment hatch is closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00068/°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°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 1.4% of instrument span.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group step counter demand position indicator shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown and control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **

ACTION:

With less than the above required group step counter demand position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each of the above required group step counter demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days.

4.1.3.3.2 OPERABILITY of the group step counter demand position indicator shall be verified in accordance with Table 4.1-1.

---

*With the Reactor Trip System breakers in the closed position.

**See Special Test Exceptions Specification 3.10.5.

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line Flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3*	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with: Steam Generator Pressure--Low	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3	15
	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	25
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
6. Auxiliary Feedwater###					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
4. Steam Line Isolation (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-High Coincident with:	21.3	2.7	0.0	$\leq$20.0 psig	$\leq$22.6 psig
Containment Pressure--High	13.3	10.3	0.0	$\leq$4.0 psig	$\leq$4.5 psig
d. Steam Line Flow--High	16.7	2.86	3.9	$\leq$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.	$\leq$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.
Coincident with: Steam Line Pressure--Low	13.0	1.16	2.3	$\geq$614 psig	$\geq$588 psig
or T_{avg}--Low	4.0	2.0	1.0	$\geq$543°F	$\geq$542.5°F
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A.	N.A.	N.A.
b. Safety Injection	see item 1			See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High (Particulate or Gaseous (See Note 1.))	1	1*	All*	Particulate $\leq 6.1\times10^{5}$CPM Gaseous See Note 2.	26 for MODES 1, 2, 3, 4 or 27 for MODES 5 AND 6
b. RCS Leakage Detection Particulate Radioactivity or Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	26
2. Spent Fuel Storage Pool Areas					
a. Unit 3 Radioactivity - High Gaseous	1	1	**	$<5.5\times10^{-2}$ $\frac{\mu Ci}{cc}$	28
b. Unit 4 Radioactivity-High Gaseous#	1	1	**	$<2.8\times10^{-2}$ $\frac{\mu Ci}{cc}$ (SPING) or $<1.0\times10^{6}$CPM (PRMS)	28

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. Containment Hydrogen Monitors	2	1	1, 2	35
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
d. Main Steam Lines	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. gen.	1/stm. gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.
2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

---

*Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

TABLE 3.3-6

FIRE DETECTION INSTRUMENTS
FOR ESSENTIAL EQUIPMENT

INSTRUMENT LOCATION FIRE ZONE AREA	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)*	FLAME (x/y)*	SMOKE (x/y)*
4 - Aux. Bldg. Corridor E. 10'			(2/0)
5 - Chem. Drain/Laundry/Shower Tank Room			(2/0)
9 - Laundry/Chemical Drain Tank Room			(1/0)
10 - Pipeway			(11/0)
11 - Unit 3 RHR Heat Exchanger Room			(5/0)
12 - RHR Pump 3A Room			(2/0)
13 - RHR Pump 3B Room			(2/0)
14 - Unit 4 RHR Heat Exchanger Room			(5/0)
15 - RHR Pump 4A Room			(2/0)
16 - RHR Pump 4B Room			(2/0)
19 - Unit 3 W Elect Penet Room			(5/0)
20 - Unit 3 S Elect Penet Room			(11/0)
21 - Instrument Shop			(2/0)
22 - Radioactive Laboratory			(2/0)
25 - Aux. Bldg. Elect. Equipmt. Room			(6/0)
25A- Spare Battery Room	(2/0)		
26 - Unit 4 N Elect Penet Room			(8/0)
27 - Unit 4 W Elect Penet Room			(6/0)
30 - Unit 4 Piping and Valve Room			(4/0)
40 - Unit 3 Piping and Valve Room			(4/0)
45 - Unit 4 Charging Pump Room	(0/4)		(3/0)
47 - Unit 4 Component Cooling Water Area	(0/4)	(5/2)	
54 - Unit 3 Component Cooling Water Area	(0/4)	(4/2)	
55 - Unit 3 Charging Pump Room	(0/4)		(3/0)
58 - Aux Bldg Corridor, El. 18'			(18/0)
59 - Unit 4 Containment Electrical Penet. Area**			(10/0)
60 - Unit 3 Containment Electrical Penet. Area**			(16/0)
61 - Reactor Control Rod Eqpmt Room - Unit 4			(4/0)
62 - Computer Room			(11/0)
63 - Reactor Control Rod Eqpmt Room - Unit 3			(4/0)
67 - 4160V Switchgear 4B			(10/0)
68 - 4160V Switchgear 4A			(6/0)
70 - 4160V Switchgear 3B			(10/0)
71 - 4160V Switchgear 3A			(6/0)
72 - Diesel Generator 3B	(0/3)	(1/0)	(1/0)
73 - Diesel Generator 3A	(0/3)	(1/0)	(1/0)
74 - Day Tank Room 3B	(1/1)		
75 - Day Tank Room 3A	(1/1)		
76 - Unit 4 Turbine Lube Oil Reservoir	(1/0)		
79A- North-South Breezeway	(0/6)		(4/0)
81 - Unit 4 Main Transformer	(1/0)		
82 - Unit 4 Aux Transformer Area	(1/0)		
84 - Unit 3 and 4 Aux Feedwater Pump Area (DC Enclosure Bldg.)			(3/0)

TABLE 3.3-6 (Continued)

FIRE DETECTION INSTRUMENTS FOR ESSENTIAL EQUIPMENT

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS		
FIRE ZONE AREA	HEAT (x/y)*	FLAME (x/y)*	SMOKE (x/y)*
87 - Unit 3 Aux Transformer Area	(1/0)		
93 - 480V Load Center 4A and 4B			(1/0)
94 - 480V Load Center 4C and 4D			(2/0)
95 - 480V Load Center 3A and 3B			(1/0)
96 - 480V Load Center 3C and 3D			(2/0)
97 - Mechanical Equipment Room			(1/0)
98 - Cable Spreading Room			(16/15)
101- RPI Inverter and MG Sets			(1/0)
102- Battery Rack 4B	(1/0)		
103- Battery Rack 3A	(1/0)		
104- RPI Inverter and MG Sets			(2/0)
106- Control Room	(1/0)		(16/0)
108A- Train A Inverters			(3/4)
108B- Train B Inverters			(4/4)
109- Battery Rack 4A	(1/0)		
110- Battery Rack 3B	(1/0)		
113- Unit 4 Feedwater Platform		(2/0)	
116- Unit 3 Feedwater Platform		(2/0)	
119- Unit 4 Intake Cooling Water Pump Area		(4/0)	
120- Unit 3 Intake Cooling Water Pump Area		(4/0)	
132- Control Room Electrical Chase			(1/2)
133- Diesel Generator 4B	(5/5)	(3/0)	(5/0)
134- 4160V Switchgear 3D Room			(2/0)
135- Diesel Generator 4B Control Panel Room			(2/0)
136- Diesel Generator 4B Fuel Transfer Pump			(2/0)
138- Diesel Generator 4A	(5/5)	(3/0)	(5/0)
139- 4160V Switchgear 4D Room			(2/0)
140- Diesel Generator 4A Control Panel Room			(2/0)
141- Diesel Generator 4A Fuel Transfer Pump			(2/0)
N/A - 18' level of the Turbine Area	(N/A)#	(N/A)#	(N/A)#

TABLE NOTATIONS

*: x is number of Function A (early warning fire dectection and notification only) instruments.
y is number of Function B (actuation of Fire Suppression Systems and early warning fire detection and notification) instruments.

** The fire detection instruments located within the containment are not required to be operable during the performance of Type A Containment Leakage Rate Test.

A fire watch patrol shall be established to inspect the 18 foot level of the Turbine Area once each hour.

TABLE 4.3-5

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluents Line	D	P	R(2)	Q(1)
b. Steam Generator Blowdown Effluent Line	D	M	R(2)	Q(1)
2. Flow Rate Measurment Devices				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
b. Steam Generator Blowdown Effluent Lines	D(3)	N.A.	R	Q

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

TABLE 3.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4.	Plant Vent System (Include Unit 4's Spent Fuel Pool)			
a.	Noble Gas Activity Monitor (SPING or PRMS)	1	*	47
b.	Iodine Sampler	1	*	48
c.	Particulate Sampler	1	*	48
d.	Effluent System Flow Rate Measuring Device	1	*	46
e.	Sampler Flow Rate Measuring Device	1	*	46
5.	Unit 3 Spent Fuel Pit Building Vent			
a.	Noble Gas Activity Monitor	1	*	47
b.	Iodine Sampler	1	*	48
c.	Particulate Sampler	1	*	48
d.	Sampler Flow Rate Measuring Device	1	*	46

TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Condenser Air Ejector Vent System (Continued)					
e. Sample Flow Rate Measuring Device	D	N.A.	R	N.A.	##
4. Plant Vent System (Include Unit 4's Spent Fuel Pool)					
a. Noble Gas Activity Monitor (SPING or PRMS)	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Measuring Device	D	N.A.	R	N.A.	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	N.A.	*

TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Unit 3 Spent Fuel Pit Building Vent					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Measuring Device	D	N.A.	R	N.A.	*

TABLE NOTATION

* At all times.
** During GAS DECAY TANK SYSTEM operation.
\# Applies during MODE 1, 2, 3 and 4.
\#\# Applies during MODE 1, 2, 3 and 4 when primary to secondary leakage is detected as indicated by condenser air ejector noble gas activity monitor.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint.

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that if the instrument indicates measured levels above the Alarm Setpoint, alarm annunciation occurs in the control room (for PRMS only) and in the computer room (for SPING only).

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

TABLE 4.3-6 (Continued)

TABLE NOTATIONS (Continued)

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal.

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER		FUNCTION
Unit 3	Unit 4	High-Head Safety Injection Check Valves
3-874A	4-874A	Loop A, hot leg
3-875A	4-875A	cold leg
3-873A	4-873A	cold leg
3-874B	4-874B	Loop B, hot leg
3-875B	4-875B	cold leg
3-873B	4-873B	cold leg
3-875C	4-875C	Loop C, cold leg
3-873C	4-873C	cold leg
		Residual Heat Removal Line Check Valves
3-876A	4-876A 4-876E	Loop A, cold leg
3-876B 3-876D	4-876B 4-876D	Loop B, cold leg
3-876C 3-876E	4-876C	Loop C, cold leg
	MOV4-750 MOV4-751	Loop A, hot leg to RHR
MOV3-750 MOV3-751		Loop C, hot leg to RHR

ACCEPTABLE LEAKAGE LIMITS

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable provided that the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previously measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between previously measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination	At least once per 72 hours.	1, 2, 3, 4
2. Tritium Activity Determination	1 per 7 days.	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
4. Radiochemical Isotopic Determination Including Gaseous Activity	Monthly	1, 2, 3, 4
5. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
6. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 µCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

SURVEILLANCE REQUIREMENTS (Continued)

d. Assuring that the observed lift-off force for each tendon exceeds the minimum required lift-off force. Required lift-off forces shall be calculated individually for each surveillance tendon prior to the beginning of each surveillance, and should consider such factors as:

   1) Prestressing history;
   2) Friction losses; and
   3) Time-dependent losses (creep, shrinkage, relaxation), considering time elapsed from prestressing.

e. Verifying the OPERABILITY of the sheathing filler grease by:

   1) Minimum grease coverage exists for the different parts of the anchorage system, and

   2) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through visual inspection that no unacceptable levels of corrosion exist on the end anchorages and no unacceptable cracking exists in the concrete adjacent to the end anchorages. Determination of acceptance levels shall be by engineering evaluation of the areas in question. If unacceptable conditions are found, the tendons inspected during the previous surveillance shall be examined to determine whether the corrosion levels or concrete cracking have increased since the previous surveillance. Inspection of adjacent concrete surfaces shall be performed concurrently with the containment tendon surveillance (Technical Specification 4.6.1.6.1).

4.6.1.6.3 Containment Surfaces. In accordance with 10 CFR 50, Appendix J. Section V. A, a visual inspection of the accessible interior and exterior surfaces of the containment, including the liner plate, shall be performed during the shutdown for (but prior to) each Type A containment leakage rate test (Technical Specificaticn 4.6.1.2). The purpose of this inspection shall be to identify any evidence of structural deterioration which may affect containment structural integrity or leaktightness. The visual inspection shall be general in nature; its intent shall be to detect gross areas of widespread cracking, spalling, gouging, rust, weld degradation, or grease leakage. The visual examination may include the utilization of binoculars or other optical devices. Corrective actions taken, and recording of structural deterioration and corrective actions, shall be in accordance with 10 CFR 50, Appendix J, Section V. A. Records of previous inspections shall be reviewed to verify no apparent changes in appearance. The first inspection performed will form the baseline for future surveillances.

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Three emergency containment filtering units shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one emergency containment filtering unit inoperable, restore the inoperable filter to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 Each emergency containment filtering unit shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release or (3) after every 720 hours of system operation by:

   1) Performance of a visual inspection for foreign material and gasket deterioration, and verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP and halogenated hydrocarbons at the system flow rate of 37,500 cfm ±10%;

   2) Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with applicable portions of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and performed in accordance with ANSI N-510-1975, meets the acceptance criteria of greater than 99.9% removal of elemental iodine; and that any charcoal failing to meet this criteria be replaced with charcoal that meets or exceeds the criteria of position C.6.a of Regulatory Guide 1.52, Rev. 2; and

   3) Verifying a system flow rate of 37,500 cfm ±10% and a pressure drop across the HEPA and charcoal filters of less than 6 inches water gauge during system operation when tested in accordance with ANSI N510-1975;

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, fire barrier penetration seals, and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware, and

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration will be inspected every 15 years.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The water level shall be maintained greater than or equal to elevation 56' - 10' the spent fuel storage pool.*

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

*The requirements of this specification may be suspended for more than 4 hours hours to perform maintenance provided a safety evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

3/4.9.12 HANDLING OF SPENT FUEL CASK

LIMITING CONDITION FOR OPERATION

3.9.12 The handling of spent fuel cask shall be limited to the following conditions:

1) The spent fuel cask shall not be moved into the spent fuel pit until all the spent fuel in the pit has decayed for a minimum of one thousand five hundred twenty-five (1,525) hours.

2) Only a single element cask may be moved into the spent fuel pit.

3) A fuel assembly shall not be removed from the spent fuel pit in a shipping cask until it has decayed for a minimum of one hundred twenty (120) days.

APPLICABILITY: During movement of spent fuel cask in the spent fuel storage area.

ACTION:

With the requirement of the above specification not satisfied, suspend all movement of the spent fuel cask within the spent fuel storage area.

SURVEILLANCE REQUIREMENTS

4.9.12.1 The following required decay times of the spent fuel assemblies shall be determined prior to the movement of a spent fuel cask by verification of date and time the spent fuel assemblies were placed into the spent fuel pit:

a. 1525 hours of decay of all spent fuel assemblies in the spent fuel pit for movement of a spent fuel cask into the spent fuel pit.

b. 120 days of decay of the spent fuel assembly in the spent fuel cask prior to removal of the spent fuel cask from the spent fuel pit.

4.9.12.2 Prior to any operations involving spent fuel cask movement into the spent fuel pit, verify only a single element cask will be moved into the spent fuel pit.

4.9.12.3 The spent fuel cask crane interlock shall be demonstrated OPERABLE within 7 days of crane operation and at least once per 7 days (7 days is maximum time between tests; specification 4.0.2 does not apply here) when the crane is being used to maneuver the spent fuel cask.

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.14 The following conditions shall apply to spent fuel storage:

a. The maximum enrichment loading for the fuel assemblies in the spent fuel racks shall not exceed 4.5 weight percent of U-235.

b. The minimum boron concentration in the Spent Fuel Pit shall be 1950 ppm.

c. Storage in Region II of the Spent Fuel Pit shall be further restricted by burnup and enrichment limits specified in Table 3.9-1.

APPLICABILITY: At all times when fuel is stored in the Spent Fuel Pit.

ACTION:

a. With either condition a, or c not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

b. With boron concentration in the Spent Fuel Pit less than 1950 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 1950 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.14 The boron concentration of the Spent Fuel Pit shall be verified to be 1950 ppm or greater at least once per month.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the GAS DECAY TANK SYSTEM (as measured in the inservice gas decay tank) shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the inservice gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the inservice gas decay tank greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the gas decay tanks and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the inservice gas decay tanks shall be determined to be within the above limits by continuously* monitoring the waste gases in the inservice gas decay tank with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-8 of Specification 3.3.3.6.

*When continuous monitoring capability is inoperable, Table 3.3-8 allows the use of grab samples.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or broad leaf vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

SPENT FUEL STORAGE (Continued)

in Region II cell locations, strict controls are employed to evaluate burnup of the spent fuel assembly. Upon determination that the fuel assembly meets the burnup requirements of Table 3.9-1, placement in a Region II cell is authorized. These positive controls assure the fuel enrichment limits assumed in the safety analyses will not be exceeded.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed to provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance in region 1 of 0.97% $\Delta k/k$ and in region 2 of 1.96% $\Delta k/k$ for uncertainties for two region fuel storage racks.

b. A nominal 10.6 inch center-to center distance for Region 1 and 9.0 inch center-to-center distance for Region 2 for two region fuel storage racks.

c. The maximum enrichment loading for fuel assemblies is 4.5 weight percent of U-235.

5.6.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.5 weight percent of U-235.

5.6.1.3 Credit for burnup is taken in determining placement locations for spent fuel in the two-region spent fuel racks. Administrative controls are employed to evaluate the burnup of each spent fuel assembly stored in areas where credit for burnup is taken. The burnup of spent fuel is ascertained by careful analysis of burnup history, prior to placement into the storage locations. Procedures shall require an independent check of the analysis of suitability for storage. A complete record of such analysis is kept for the time period that the spent fuel assembly remains in storage onsite.

DRAINAGE

5.6.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1404 in two region storage racks

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

6.1 RESPONSIBILITY

6.1.1 The Plant Manager - Nuclear shall be responsible for overall unit operation of both units and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Plant Supervisor - Nuclear (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for facility operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to, and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Topical Quality Assurance Report and updated in accordance with 10 CFR 50.54(a)(3).

b. The President-Nuclear Division shall have corporate responsibility for overall plant nuclear safety, and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

c. The Plant Manager-Nuclear shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the health physics manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

RESPONSIBILITIES (Continued)

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board;

f. Review of all REPORTABLE EVENTS;

g. Review of reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment or systems that affect nuclear safety.

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager - Nuclear or the Chairman of the Company Nuclear Review Board;

i. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Chairman of the Company Nuclear Review Board;

j. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL;

k. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board.

6.5.1.7 The PNSC shall:

a. Recommend in writing to the Plant Manager - Nuclear approval or disapproval of items considered under Specification 6.5.1.6a. through d. prior to their implementation and items considered under Specification 6.5.1.6i through k.

b. Provide written notification within 24 hours to the Plant Manager-Nuclear, President-Nuclear Division and the Company Nuclear Review Board of disagreement between the PNSC and the Plant Manager-Nuclear; however, the Plant Manager - Nuclear shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

PROCEDURES AND PROGRAMS (Continued)

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed in accordance with Specification 6.5.3 and approved by the Plant Manager-Nuclear or the department head of the responsible department within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the Safety Injection System, Chemical and Volume Control System, and the Containment Spray System. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

PROCEDURES AND PROGRAMS (Continued)

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis, and

(3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of the following year.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls, as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required by Specification 3.12.3; reasons for not conducting the Radiological Environmental Monitoring Program as required by specification 3.12.1, and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

RADIATION PROTECTION PROGRAM (Continued)

maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Health Physics Shift Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO. 144 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 28, 1991, Florida Power and Light Company (the licensee) requested approval of amendments to the Technical Specifications (TS) for the Turkey Point Plant, Units 3 and 4. Specifically, the proposed amendments would revise the TS by removing outdated material, incorporating administrative clarifications, and correcting typographical errors. These changes represent an administrative upgrade to the Turkey Point Units 3 and 4 TS.

2.0 EVALUATION

2.1 TS 1.0 - Definitions

On page 1-2, TS 1.7 a.2), the reference to Table 3.6-1 is deleted. This change corrects a typographical error related to referencing Table 3.6-1, which does not exist in the TS. Specification 3.6.4 meets the intent of the exception statement. The NRC staff has reviewed this change and finds it acceptable.

2.2 TS 2.0 - Safety Limits and Limiting Safety System Settings

On page 2-10, Table 2.2-1, K6 definition, the words "and K6" are deleted and the portion of the definiton where the words "= 0 for T < T" appear are moved to the following line. The expression is modified to ensure a consistent mathematical convention. The staff finds this change acceptable.

2.3 TS 3/4.1 - Reactivity Control Systems

On page 3/4 1-23, TS 4.1.3.3.2, the sentence "A CHANNEL CHECK CALIBRATION AND ANALOG CHANNEL OPERATIONAL TEST shall be performed per Table 4.1-1" is deleted and the wording "OPERABILITY of the group step counter demand position indicator shall be verified in accordance with Table 4.1-1" is substituted.

This change is an administrative clarification. Table 4.1-1 establishes operability of the group step counter demand position indicator by performing a CHANNEL CHECK and an OPERATIONAL TEST. The demand position indicator system is a digital system and does not have channels to calibrate or perform an analog operational test. This change is consistent with the surveillance requirements of Table 4.1-1. The staff finds this change acceptable.

On page 3/4 1-23, TS 3.1.3.3, Footnote (**), the reference for the "Special Test Exceptions Specification 3.10.4" is changed to read "See Special Test Exceptions Specification 3.10.5".

This corrects a typographical error. Special Test Exceptions Specification 3.10.4 does not exist. The staff finds this change to be acceptable.

2.4 TS 3/4.3 - Instrumentation

On page 3/4 3-15, Table 3.3-2, functional unit 1.f., Steam Generator Pressure--Low, under column heading CHANNELS TO TRIP, the licensee requested that the word "generator" is substituted for the word "line". This is an administrative clarification. The steam generator low pressure system actuation signal is monitored with one channel per steam generator. The column headings CHANNELS TO TRIP and MINIMUM CHANNELS OPERABLE are established based upon the operability of this one channel per steam generator. Substitution of the word "generator" for the word "line" is made to ensure a consistent interpretation of the operability criteria based upon the total number of channels per steam generator. The staff find this change to be acceptable.

The licensee also requested a TS change on Table 3.3-2, page 3-15 for the Steam Generator Pressure - Low Signal column for CHANNEL TO TRIP and MINIMUM CHANNELS OPERABLE, which would have inserted the word "generators" in two places for the word "lines". The staff has denied this request to be consistent with the interpretation of the actual location of the pressure transmitters (PTs), which read the steam generator pressure signal downstream from the steam generator in the two steam lines. This interpretation of PT location is also consistent with the wording used in NUREG-0452, Westinghouse Standardized Technical Specifications.

On page 3/4 3-18, Table 3.3-2, functional unit 4.d., Steam Line Flow - High, the wording "in any two steam lines" is inserted under the columns CHANNELS TO TRIP and MINIMUM CHANNELS OPERABLE. The steam line high flow system protection is based upon the presence of two channels per steam line. A review of the Turkey Point Plant, Units 3 and 4 control system diagram for steam line break protection confirms that the logic for CHANNELS TO TRIP is based on one channel per steam line in any two steam lines where it is assumed that one of the flow transmitters in one of the three steam lines has failed and its signal is placed in the trip condition. This channel logic is consistent with the channel logic criteria as specified in NUREG-0452. The addition of the wording "in any two steam lines" ensures consistency with this NUREG. The staff finds that this change is acceptable.

The licensee also requested a TS change on Table 3.3-2, page 3/4 3-18 for the Steam Line Isolation - Steam Generator Pressure - Low columns for CHANNELS TO

TRIP and MINIMUM CHANNELS OPERABLE, which would have inserted the word "generators" in two places for the word "lines". The staff has again denied this request to be consistent with the interpretation of the actual location of the pressure transmitters (PTs) which read the steam generator pressure signal downstream from the steam generator in the two steam lines. This interpretation of PT location is also consistent with the wording used in NUREG-0452, Westinghouse Standardized Technical Specifications.

On page 3/4 3-26, Table 3.3-3, the letter "f" in 4.f. is changed to the letter "d". A review of NUREG-0452 confirms that no information was unintentionally deleted and that this change is simply a typographical correction.

On page 3/4 3-36, Table 3.3-4, the word MINIMUM is inserted in the heading CHANNELS OPERABLE, and the word "for" is added between the word "27" and "MODES" in Functional Unit 1.a., Containment Atmosphere Radioactivity - High.

These were typographical errors. Inserting the word "MINIMUM" in the colunm heading CHANNELS OPERABLE ensures a consistency with ACTION statements 27 and 28 of Table 3.3-4. The addition of the word "for" provides a clarification of the applicability of ACTION statement 27 for Modes 5 and 6. The NRC staff finds these changes acceptable.

On page 3/4 3-43, Table 3.3-5, the word "quadrant" is inserted after the word "core" under the column MINIMUM CHANNELS OPERABLE for Instrument 14, In Core Thermocouples (Core Exit Thermocouples).

This corrects a typographical error. The addition of the word "quadrant" ensures consistency with the definition for the TOTAL NO. OF CHANNELS for Instrument 14. This wording is identical to the wording in Table 3.3-10 of NUREG-0452, Rev. 4, Standard Technical Specifications for Westinghouse Pressurized Water Reactors. The staff has reviewed this change and finds it acceptable.

On pages 3/4 3-48 and 3/4 3-49, Table 3.3-6, under the column heading TOTAL NUMBER OF INSTRUMENTS, an asterisk (*) is added so that the heading reads "(x/y)*" for the HEAT, FLAME and SMOKE instruments. Under TABLE NOTATIONS, the expression "(x/y)" is deleted and the location of the asterisk is centered.

These were typographical errors. The definition for (x/y) is identical for the three types of instruments (heat, flame, and smoke) and was never intended to be distinguished as any different. Adding the asterisk for each instrument ensures the correct and identical definition for each type of fire detection instrument. The NRC staff finds these changes acceptable.

On page 3/4 3-53, Table 4.3-5, the asterisk ("*") footnote, which states that the "Channel calibration frequency shall be at least once per 18 months" is deleted since the definition for "R" as defined in Table 1.1 is identical.

This is an administrative clarification. Deleting the asterisk footnote, while maintaining the notation "R", reduces redundancy while maintaining consistency within the TS. The staff finds this change acceptable.

On page 3/4 3-53, Table 4.3-5, TABLE NOTATIONS (1), second line, the word "measured" is substituted for the word "measures".

This was a typographical error related to the use of an incorrect verb syntax. The staff finds this change acceptable.

On page 3/4 3-53, TABLE NOTATIONS (2), the reference standards certifier name of "National Bureau of Standards (NBS)" is changed to "National Institute of Standards and Technology (NIST)."

This is an administrative clarification which reflects a change in the title of a government agency. The staff finds this change acceptable.

On page 3/4 3-56, Table 3.3-8, 4.d., the values under the three columns are moved over several spaces to line-up with the values in Table 3.3-8.

This was a strictly editorial typographical error. The staff finds this change acceptable.

On pages 3/4 3-59 and 3/4 3-61, Table 4.3-6, footnote "(6)" on page 3/4 3-61 is deleted, and "R" is substituted for "(6)" under the column heading CHANNEL CALIBRATION for Instruments 4.a., 4.d., and 4.e. on page 3/4 3-59.

These are administrative clarifications. These changes reduces redundancy, since the definition for "R" as provided in Table 1.1 is identical to the definition for "(6)" as defined in Table 4.3-6. The staff finds these changes acceptable.

On page 3/4 3-60, TABLE NOTATIONS (3), the reference standards certifier name of "National Bureau of Standards (NBS)" is change to "National Institute of Standards and Technology (NIST)."

This is an administrative clarification which reflects a change in the title of a government agency. The staff finds this change acceptable.

2.5 TS 3/4.4 - Reactor Coolant System (RCS)

On page 3/4 4-22, Table 3.4-1, the valve for Unit 3 should read "3-876A", as opposed to "3-876-A". Reactor coolant system pressure isolation valves MOV3-750 and MOV3-751 are incorrectly listed as valves in the Loop A hot leg to the residual heat removal (RHR) system. These valves are actually located in the Loop C hot leg to the RHR. Table 3.4-1 is revised to reflect the correct location of MOV3-750 and MOV3-751.

These changes ensure accuracy with the plant configuration. For each unit, the RCS pressure isolation valve is located on at least one hot leg to the RHR. TS 3.4.6.2e. states that RCS leakage shall be limited to a maximum of 5 gpm from any reactor coolant system pressure isolation valve. This change identifies the actual loop the Unit 3 motor-operated valves in the hot leg to the RHR are located, without compromising the system protection. The staff has reviewed these changes and find them acceptable.

On page 3/4 4-28, Table 4.4-4, the word "AND" for the word "ANY" is substituted in the column heading "TYPE OF MEASUREMENT ANY ANALYSIS".

This corrects a typographical error and ensures accuracy and consistency with NUREG-0452, Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Rev. 4. The staff finds this change acceptable.

2.6 TS 3/4.6 - Containment Systems

On page 3/4 6-10, TS 4.6.1.6.3, fifth line down, the reference for TS "4.6.1.2.1" is changed to "4.6.1.2".

TS 4.6.1.2.1. does not exist in the TS. Eliminating the ".1", at the end of this TS ensures that the intent of this cross-reference is maintained. The staff has reviewed this change and finds it acceptable.

On page 3/4 6-15, TS 4.6.3.b2), a period between "6" and "a" is inserted in the expression "criteria of position C.6a".

This corrects a typographical error. The staff has reviewed this change and finds it acceptable.

2.7 TS 3/4.7 - Plant Systems

On page 3/4 7-33, TS 3.7.9, the wording "fire barrier penetration seals" is relocated from an example of "fire rated assemblies" to an example of "all sealing devices in fire rated assembly penetrations."

This is an administrative clarification. TS 4.7.9.1 states that at least once per 18 months the required fire rated assemblies and penetration sealing devices shall be verified operable. The surveillance requirement differentiates the extent of the inspection based upon the category of the equipment (i.e., fire rated assemblies versus penetration sealing devices). By substituting the wording "fire barrier penetration seals" as an example of sealing devices in fire rated assembly penetrations, the surveillance requirements are consistent with TS 3.7.9. The staff finds this change acceptable.

2.8 TS 3/4.9 - Refueling Operations

On page 3/4 9-12, TS 3.9.11, the asterisk footnote which reads "During spent fuel rerack operation, the water level may be lowered to a level justfied by an engineering safety evaluation. There will be no movement of fuel assemblies with water level lower than 56' - 10" elevation during rerack operation" is deleted. Also, the asterisk in the second line of TS 3.9.11 is deleted. All double asterisk notations are changed to a single asterisk notation.

This is an administrative clarification. This footnote is no longer applicable, since both Turkey Point Units 3 and 4 spent fuel pools have been reracked with two region high-density spent fuel storage racks. The staff finds this change acceptable.

On page 3/4 9-13, TS 3.9.12, the asterisk footnote, which reads "The spent fuel cask can be moved into the Unit 4 spent fuel pit after a minimum decay of 1000 hours until the new two-region high density spent fuel racks are installed" is deleted. The asterisk in TS 3.9.12 1) is also deleted.

This footnote is no longer applicable, since both Turkey Point Units 3 and 4 spent fuel pools have been reracked with two-region high density spent fuel storage racks. The staff finds this change acceptable.

On page 3/4 9-13, the TS number "4.9.12.3" is substituted for the number "4.8.12.3".

This corrects a typographical error. The number "8" was inadvertently substituted for the number "9" in the specification number. The staff finds this change acceptable.

On page 3/4 9-15, TS 3/4.9.14, Spent Fuel Storage, is modified to reflect the fact that Turkey Point Units 3 and 4 spent fuel pools have been reracked with two-region high density spent fuel storage racks. The following changes are requested:

(a) The wording in TS 3.9.14a., which reads "Fuel Assemblies containing more than 4.1 weight percent of U-235 shall not be placed in the single region spent fuel storage racks. After installation of the two region high density spent fuel racks,..." is deleted.

(b) TS 3.9.14d. is deleted.

(c) In ACTION statement a., the wording "either condition a or c" is substituted for the wording "any of conditions a, c or d", and

(d) The asterisk footnote, which reads "These requirements are applicable only after installation of the new two-region high density spent fuel racks" is deleted. Also, the asterisk in TS 3.9.14c. is deleted.

These changes are no longer applicable, since both Turkey Point Units 3 and 4 spent fuel pools have been reracked with two-region high density spent fuel storage racks. The NRC staff finds these changes acceptable.

On page 3/4 9-15, TS 3.9.14a., the wording "not exceed" is substituted for the word "be".

This is an administrative clarification. The intent of this TS is to establish an upper-limit on the fuel assembly enrichment in the spent fuel pool. The insertion of this wording ensures a consistent interpretation for the limiting condition for operation for the spent fuel storage racks and does not change the actual enrichment limit. The staff finds this change acceptable.

2.9 TS 3/4.11 - Radioactive Effluents

On page 3/4 11-15, TS 4.11.2.5, the reference to "Table 3.3-8 of Specification 3.3.3.7" is substituted for "Table 3.3-8 of Specification 3.3.3.6". Also, in the asterisk footnote, the reference from "Table 3.3-9" is changed to "Table 3.3-8".

These are administrative clarifications. The corrections are being made to ensure compliance with the appropriate TS. Both TS 3.3.3.7 and Table 3.3-9 do not exist in the TS. TS 3/4.11.2.5 applies to the Gas Decay Tank System, while TS 3.3.3.6 applies to "Radioactive Gaseous Effluent Monitoring Instrumentation", and therefore the correct cross-reference is TS 3.3.3.6. The staff finds these changes acceptable.

The asterisk footnote provided on page 3/4 11-15 addresses the inoperability of the continuous monitoring capability for concentrations of hydrogen and oxygen. ACTION statement 49, in Table 3.3-8, clearly addresses this condition. The staff has reviewed these changes and finds them acceptable.

2.10 TS 3/4.12 - Radiological Environmental Monitoring

On page 3/4 12-2, TS 3.12.1, ACTION statement c, the wording "broad leaf" is substituted for the wording "fresh leafy".

This is an administrative clarification. The correct terminology is "broad leaf", as used in Table 3.12-1 and the associated TABLE NOTATION (11). This substitution ensures consistency within the reporting requirements. The NRC staff finds this change acceptable.

2.11 Bases for Sections 3.0 and 4.0

On page B 3/4 9-4, the footnote "*" and the corresponding asterisk in the text are deleted.

These are administrative clarifications. This footnote is no longer required, since the actual configuration of the Turkey Point Units 3 and 4 spent fuel pool is the two-region high density spent fuel racks. As a result, the statement which referenced this footnote is governed by the TS. The NRC staff finds these changes acceptable.

2.12 TS 5.0 - Design Features

On page 5-5, TS 5.6.1, Criticality, is modified to reflect the fact that the Turkey Point Units 3 and 4 spent fuel pool was reracked with two-region high density spent fuel storage racks. The following changes are requested:

(a) TS 5.6.1.1a. and 5.6.1.1d. are deleted.

(b) In TS 5.6.1.1c., the first sentence which reads "A nominal 13.7 inch center-to-center distance between fuel assemblies placed in the single-region storage racks" is deleted.

(c) In TS 5.6.1.1e., the beginning of the first sentence which reads "After installation of the two-region high density spent fuel storage racks,..." is deleted.

These are administrative clarifications. These TS can be deleted, since both Turkey Point Units 3 and 4 spent fuel pools have been reracked with two-region high density spent fuel storage racks and the references to the "single region spent fuel storage racks" are no longer applicable. The NRC staff finds these changes acceptable.

On page 5-6, TS 5.6.1.3, the asterisk footnote which reads "During rack installation, it will be necessary to temporarily store Region I fuel in the Region II spent fuel racks. Administrative controls will be utilized to maintain a checkerboard storage configuration, i.e., alternate cell occupation, in the Region II racks" is deleted. Also, the asterisk in the second line of TS 5.6.1.3 is deleted.

This is an administrative clarification. This footnote can be deleted, since both Turkey Point Units 3 and 4 spent fuel pools have been reracked with two-region high density spent fuel storage racks. Reference to the "temporary storage of Region I fuel in the Region II spent fuel racks" is no longer applicable. The NRC staff finds this change acceptable.

On page 5-6, TS 5.6.3, the statement "... 621** fuel assemblies in one region storage racks or .." is deleted. Also, footnote "**" is deleted.

This is an administrative clarification. The statement and footnote can be deleted, since both Turkey Point Units 3 and 4 spent fuel pools have been reracked with two-region high density spent fuel storage racks and the reference to the maximum number of fuel assemblies in one region storage racks is no longer applicable. The NRC staff finds this change acceptable.

2.13 TS 6.2 - Organization

On page 6-1, TS 6.2.1a., third line, the wording from "intermediate levels to <u>an</u> including" is changed to the wording "intermediate levels to, <u>and</u> including". (A comma is added between the words "to" and "and".)

These changes correct typographical errors by clarifying the intent of the line of authority, responsibility and communications. The NRC staff finds this change acceptable.

On page 6-7, TS 6.5.1.7b., third line, "-" is inserted, between the words "Manager" and "Nuclear", such that the wording reads "Plant Manager - Nuclear".

This corrects a typographical error. This correction is made to ensure consistency within TS 6.0. The NRC staff finds this change acceptable.

On pages 6-14 and 6-15, TS 6.8.4, the title "c. <u>Secondary Water Chemistry</u>" is moved from the top of page 6-15 to the bottom of page 6-14.

This corrects a typographical error. The title is moved to maintain consistency with the implied intention. This change parallels NUREG-0452, Rev. 4, Standard Technical Specifications for Westinghouse Pressurized Water Reactors. The staff finds this change acceptable.

On page 6-17, an asterisk is added to the heading "Annual Radiological Environmental Operating Report". In the fourth paragraph, second line, an additional asterisk is added to the asterisk after the words "two legible maps".

These were typographical errors. These changes ensure consistency between the footnotes and the statements associated with the asterisks. The NRC staff finds these changes acceptable.

On page 6-22, the page heading "Record Retention" is changed to read "Radiation Protection Program".

This change is consistent with TS conventions and the section name on the bottom of page 6-21. The staff finds this change acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to 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 (56 FR 31434). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These amendments also involve changes in administrative procedures and reporting requirements. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth under 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

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.

Principal Contributor: F. Talbot

Date: September 25, 1991

UNITED STATES NUCLEAR REGULATORY COMMISSION

FLORIDA POWER AND LIGHT COMPANY

DOCKET NOS. 50-250 AND 50-251

NOTICE OF DENIAL OF PORTION OF APPLICATION FOR

AMENDMENTS TO FACILITY OPERATING LICENSES

AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) has denied a portion of a request by Florida Power and Light Company (licensee) for amendments to Facility Operating License Nos. DPR-31 and DPR-41, issued to the licensee for operation of the Turkey Point Plant, Unit Nos. 3 and 4, located in Dade County, Florida. Notice of Consideration of Issuance of the amendments was published in the FEDERAL REGISTER on July 10, 1991 (56 FR 31434).

The purpose of the licensee's amendment request was to revise the Technical Specifications (TS) by (1) removing outdated material, (2) incorporating administrative changes, and (3) correcting typographical errors.

The NRC staff has concluded that two requested changes cannot be granted. The licensee was notified of the Commission's denial of the two proposed changes by letter dated September 25, 1991.

By November 5, 1991, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC., by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Harold F. Reis, Esq., Newman and Holtzer, P.C., 1615 L Street, NW, Washington DC 20036, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendments dated May 28, 1991, and (2) the Commission's letter to the licensee dated September 25, 1991.

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the Environmental and Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of Item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Document Control Desk.

Dated at Rockville, Maryland, this 25th day of September 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Rajender Auluck

Rajender Auluck, Acting Director
Project Directorate II/2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation