October 17, 1984

See Correction letter of 12/13/84

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Docket Nos. 50-250
and 50-251

Mr. J. W. Williams, Jr., Vice President
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Williams:

The Commission has issued the enclosed Amendment No. 110 to Facility Operating License No. DPR-31 and Amendment No. 104 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 June 15, 1984. However, by letter dated September 13, 1984, you withdrew the request for technical specifications related to the Reactor Vessel Level Monitoring System (RVLMS). The RVLMS is a portion of the Inadequate Core Cooling Instrumentation (ICCI) which is addressed in Item 10 of the Generic Letter. The NRC staff has not completed their review of the ICCI for the Turkey Point Plant and will not issue the proposed technical specifications for Item 10 until the review is completed.

These amendments revise the Technical Specifications for the TMI Items in response to our Generic Letter 83-37, dated November 1, 1983, as follows:

1. Reactor Coolant System Vents (II.B.1)
2. Post-Accident Sampling (II.B.3)
3. Long Term Auxiliary Feedwater System Evaluation (II.E.1.1)
4. Noble Gas Effluent Monitors (II.F.1.1)
5. Sampling and Analysis of Plant Effluents (II.F.1.2)
6. Containment High-Range Radiation Monitor (II.F.1.3)
7. Containment Pressure Monitor (II.F.1.4)
8. Containment Water Level Monitor (II.F.1.5)
9. Containment Hydrogen Monitor (II.F.1.6)

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

/s/DMcDonald

Daniel G. McDonald, Jr., Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No.110 to DPR-31
2. Amendment No.104 to DPR-41
3. Safety Evaluation

cc: w/enclosures
See next page

Hold until check for comments on 9/21

ORB#1:DL	ORB#1:DL	C-ORB#1:DL	OELD	AD:OR:DL
CParrish;ps	DMcDonald;ps	SVarga	MYoung	GLainas
8/22/84	8/23/84	8/2?/84	9/11/84	10/1/84

J. W. Williams, Jr.
Florida Power and Light Company

Turkey Point Plants
Units 3 and 4

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UNITED STATES
NUCLEAR REGULATORY COMM SSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT C MPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT 0. 3

AMENDMENT TO FACILITY OPERATI G LICENSE

Amendment No. 110
License No. DPR-31

1. The Nuclear Regulatory Commission (the Commi sion) has found that:

   A. The application for amendment by Floric Power and Light Company (the licensee) dated June 15, 1984, com lies with the standards and requirements of the Atomic Energy Act c 1954, as amended (the Act) and the Commission's rules and regulati ns set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rule and regulations of the Commission;

   C. There is reasonable assurance (i) that he activities authorized by this amendment can be conducted witt ut endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commis ion's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health a d safety of the public; and

   E. The issuance of this amendment is in ac ordance with 10 CFR Part 51 of the Commission's regulations and ll applicable requirements have been satisfied.

2. Accordingly, the license is amended by chang s to the Technical Specifications as indicated in the attachmer to this license amendment, and paragraph 3.B of Facility Ope ating License No. DPR-31 is hereby amended to read as foll ws:

(B) Technical Specifications

The Technical Specifications contained in Appendix A and B, as revised through Amendment No. 110 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 17, 1984

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. 104
License No. DPR-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated June 15, 1984, 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

The Technical Specifications contained in Appendix A and B, as revised through Amendment No. 104 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 17, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 110 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 104 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
i, iii and iv	i, iii and iv
1-7	1-7
3.1-1	3.1-1
3.1-1a	3.1-1a
-	3.1-1b
3.1-2	3.1-2
-	3.1-2a
3.1-3	3.1-3
3.1-4	3.1-4
3.1-5	3.1-5
3.1-5a	3.1-5a
3.1-5b	-
3.1-6	3.1-6
3.1-7	3.1-7
3.5-1	3.5-1
Table 3.5-5	Table 3.5-5
-	Table 3.5-5 (continued)
3.8-1	3.8-1
3.8-2	3.8-2
Table 4.1-1 Sheet 2	Table 4.1-1 Sheet 2
Table 4.1-1 Sheet 3	Table 4.1-1 Sheet 3
Table 4.1-1 Sheet 4	Table 4.1-1 Sheet 4
-	Table 4.1-1 Sheet 5
4.7-3	4.7-3
4.7-4	4.7-4
-	4.7-5
4.19-1	4.19-1
6-30	6-30
B3.1-1	B3.1-1
B3.1-1a	B3.1-1a
B3.1-2	B3.1-2
-	B4.19-1

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AMENDMENT NOS. 110 and 104

1.24 $\overline{E}$ - AVERAGE DISINTEGRATION ENERGY

$\overline{E}$ 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 (in MeV) for isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95% of the total noniodine activity in the coolant.

3.1 REACTOR COOLANT SYSTEM

Applicability: Applies to the operating status of th Reactor Coolant System.

Objective: To specify those limiting conditions or operation of the Reactor Coolant System which must be met to assure ;afe reactor operation.

Specification: 1. OPERATIONAL COMPONENTS

a. Reactor Coolant Pumps

1. A minimum of ONE pı np shall be in operation when the reactor is in power operε ion, except during low power physics tests.

2. A minimum of ONE pu ıp, or ONE Residual Heat Removal Pump, shall be in ope :tion during reactor coolant boron concentration reduction.

3. Reactor power shall not exceed 10% of rated power unless at least TWO reactor coola pumps are in operation.

4. Reactor power shall not exceed 45% of rated power with only two pumps in operation unless the overtemperature ΔT trip setpoint, K_1, for two loo operation, has been set at 0.88.

5. A reactor coolant pum shall not be started when cold leg temperature is $\leq 275^{o}F$ unless steam generator secondary water temperature is less than $50^{o}F$ above the RCS temperature (including i trument error).

b. Steam Generators

1. A minimum of TWO ste n generators shall be operable when the average coolant tem erature is above $350^{o}F$.

c. Pressurizer Safety Valves

1. ONE valve shall be operable whenever the head is on the reactor vessel except during hydrostatic test.

2. THREE valves shall be operable when the reactor coolant average temperature is above 350°F or the reactor is critical.

d. Pressurizer

The pressurizer shall be operable with a steam bubble, and with at least 125 KW of pressurizer heaters capable of being supplied by emergency power, when the reactor coolant is heated above 350°F.

e. Relief Valves

1. A power operated relief valve (PORV) and its associated block valve shall be operable when the reactor coolant is heated above 350°F.

2. If the average coolant temperature is greater than 350°F and the conditions of 3.1.1.e.1 cannot be met because one or more PORV(s) is inoperable, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in a condition with Keff < 0.99 within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. If the average coolant temperature is greater than 350°F and the conditions of 3.1.1.e.1 cannot be met because one or more block valve(s) is inoperable, within 1 hour either restore the block valve(s) to operable status or close the block valve(s) and remove power from the block valve(s); otherwise, be in a condition with Keff < 0.99 within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. Reactor Coolant System Vents

1. At least one reactor coolant system vent path consisting of at least two valves in series powered from emergency busses shall be OPERABLE and closed at each of the following locations when T_{avg} is greater than 200°F:

   a. Reactor Vessel Head

   b. Pressurizer steam space

2. With one of the above reactor coolant system vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. At power Operation. With both reactor coolant system vent paths inoperable, maintain the inoperable vent paths closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. At Subcritical Conditions and $T_{avg} > 200°F$. With both reactor coolant system vent paths inoperable, maintain the inoperable vent paths closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the following 30 hours.

2. PRESSURE-TEMPERATURE LIMITS

The Reactor Coolant System (except for the pressurizer) pressure a d temperature shall be limited during heatup, cooldown, criticali y (except for lower power physics tests), and inservice leak a d hydrostatic testing in accordance with the limit lines shown n Figures 3.1-1a through 3.1-1d (Unit 3) and 3.1-2a through 3.1-2d (Ur t 4). Allowable pressure-temperature combinations are BELOW AN D TO THE RIGHT of the lines on the Figures. Heatup and cooldo n rate limits are:

a. A maximum heatup rate of 100°F in any one hour.

b. A maximum cooldown rate of 100°F in any one hour.

c. A maximum temperature change of $\geq 5^{\circ}$F in any one hour duri g hydrostatic testing operation above system design pressure.

The pressurizer pressure and temperature shall be limited n accordance with the following:

d. The pressurizer shall be limited to a maximum heatup rate f 100°F in any one hour, and a maximum cooldown rate of 200°F n any one hour.

e. The pressurizer shall be limited to a maximum Reactor Coola t System spray water temperature differential of 320°F.

With any of the above limits exceeded, restore the temperatu e and/or pressure within the limits within 30 minutes, determine th t the RCS or pressurizer remains acceptable for continued operatio s or, if at power, be in at least hot shutdown within the next 6 hou s and cold shutdown within the following 30 hours.

With reactor power less than 70 percent Rated Thermal Power, the moderator temperature coefficient* shall not be more positive than $+5 \times 10^{-5}$ $\Delta K/K/^{o}F$. When this condition is not met, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization and cooldown.

With reactor power greater than or equal to 70 percent Rated Thermal Power, the moderator temperature coefficient shall not be more positive than 0 $\Delta K/K/^{o}F$. When this condition is not met, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization and cooldown.

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* These moderator temperature coefficient conditions do not apply to low power physics tests.

3. LEAKAGE

a. Any reactor coolant system leakage indiciation in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 hours of the indication (ex. water inventory changes, radiation level increases, visual or audible indication). A leak shall be assumed to exist until it is determined that no unsafe condition exists and that the indicated leak cannot be substantiated. Leakage of reactor coolant through reactor pump seals and system valves to connecting closed systems from which coolant can be returned to the reactor coolant system shall not be considered as leakage except that such losses shall not exceed 30 gpm.

b. If a reactor coolant system leakage indication is proven real, and is not evaluated as safe, or exceeds 10 gpm, reactor shutdown shall be initiated within 24 hours of the initial indication, except as noted in Section 3.1.3.g.

c. If reactor coolant leakage exists through a fault in the system boundary that cannot be isolated (ex. vessels, piping, valve bodies) the reactor shall be shutdown, and cool down to cold shutdown shall be initiated within 24 hours.

d. The safety evaluation shall consider the source and magnitude of the leak, rates of change of detection variables, and if shutdown is required, this evaluation shall be used to determine shutdown rates and conditions. A written log of the action taken shall be made as soon as practicable. The evaluation shall assure that no potential gross leak is developing and that potential release of activity will be within the guidelines of 10 CFR 20.

e. After shutdown, corrective action shall be taken before operation is resumed.

f. Above 2% of rated power, two leak detection systems of different principles shall be operable one of which is sensitive to radioactivity. The latter may be out of service for 48 hours provided two other systems are operable.

g. Reactor Coolant System leakage shall be limited to 1 gpm total primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System.

4. MAXIMUM REACTOR COOLANT ACTIVITY

The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries per gram.

With the above limits being exceeded, the following actions shall be taken:

1. When the reactor is critical or average reactor coolant temperature is greater than 500°F:

   a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.1-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.3 within 30 days indicating the number of hours above this limit.

b. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.1-1, be in a SHUTDOWN condition with average reactor coolant temperature less than 500 F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\overline{E}$ microcuries per gram, be in a SHUTDOWN condition with average reactor coolant temperature less than 500 F within 6 hours.

2. <u>For all modes of operation</u>

a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/\overline{E}$ microcuries per gram, perform the sampling and analysis requirements of item 1.h.1 of Table 4.1-2 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.b. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

5. REACTOR COOLANT CHEMISTRY

a. The following are reactor coolant chemistry concentration maximum limits in ppm when coolant is above 250°F.

	Normal Limit	Transient Limit
Oxygen	0.10	1.0
Chloride	0.15	1.5
Fluoride	0.15	1.5

b. Corrective action shall be initiated if a normal limit is exceeded or if it is anticipated from trends that normal limits may be exceeded.

c. Cold shutdown shall be initiated if the transient limits are reached, or if the corrective action required in (b) above is ineffective in reducing transient concentrations within 24 hours. If the maximum concentration of any of the elements listed did not exceed the listed transient value, operation may be resumed after corrective action has been taken. Otherwise, a safety review shall be made prior to startup.

d. When reactor coolant is 250°F or below the limits in (a.) above will be maintained, except coolant oxygen content will reach saturation conditions during refueling operations. If these limits are exceeded, the unit will be brought to cold shutdown and corrective action taken.

e. Reactor coolant pump operation shall be permitted to ensure mixing during the corrective action phases specified above and shall be permitted at temperatures 50°F above the normal cold shutdown limit when bringing the reactor to cold shutdown or after reaching cold shutdown.

6. DNB PARAMETERS

The following DNB related parameter limits shall be maintained during power operation.

a. Reactor Coolant System Tavg $\leq$ 578.2 F

b. Pressurizer Pressure $\geq$ 2220 psia*

c. Reactor Coolant Flow $\geq$ 268,500 gpm

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limits after each refueling cycle.

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

This amendment effective as of date of issuance for Unit 3 and date of Start-up, Cycle 10, for Unit 4.

3.5 INSTRUMENTATION

Applicability: Applies to reactor safety and features and accident monitoring instrumentation systems.

Objective: To delineate the conditions of the instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

1. Tables 3.5-1 through 3.5-5 state the minimum instrumentation operation conditions. Specification 3.0.1 applies to Tables 3.5-1 through 3.5-3.

TABLE 3.5-5

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENTATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE ACTIONS
1.	Pressurizer Water Level	2	1	1,2
2.	Auxiliary Feedwater Flow Rate	2 per generator	1 per generator	1,2
3.	Reactor Coolant System Subcooling Margin Monitor	2	1	1,2
4.	PORV Position Indicator (Primary Detector)	1/valve	1/valve	4
5.	PORV Block Valve Position Indicator	1/valve	1/valve	4
6.	Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve	1,2
7.	Containment Pressure (Wide Range)	2	1	1,2
8.	Containment Pressure (Narrow Range)	2	1	3
9.	Containment Water Level (Wide Range)	2	1	1,2
10.	Containment Water Level (Narrow Range)	2	1	3
11.	Containment High Range Area Radiation	2	1	5
12.	Containment Hydrogen Monitors	2	1	6,7
13.	High Range - Noble Gas Effluent Monitors			
	a. Plant Vent Exhaust	1	1	5
	b. Unit 3 - Spent Fuel Pit Exhaust	1	1	5
	c. Condenser Air Ejectors	1	1	5
	d. Main Steam Lines	1	1	5
14.	Incore Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1,2

TABLE 3.5-5 (Continued)

ACTION STATEMENTS

ACTION 1 With the number of OPERABLE accident monitoring instrumentati channel(s) less than the Total Number of Channels shown in Table 3.5-5. eith restore the inoperable channel(s) to OPERABLE status within 7 days, or be in condition with $K_{eff} < 0.99$, % thermal power excluding decay heat equal zero, and an average coolant temperature $T_{avg} < 350^{o}F$ within the next hours.

ACTION 2 With the number of OPERABLE accident monitoring instrumentation channe less than the minimum channels OPERABLE requirements of Table 3.5- either restore the inoperable channel(s) to OPERABLE status within 48 hour or be in a condition with $K_{eff} < 0.99$, % thermal power excluding decay he equal to zero, and an average coolant temperature $T_{avg} < 350^{o}F$ within tl next 12 hours.

ACTION 3 Operation may continue up to 30 days with less than minimum channe OPERABLE for narrow range instruments.

ACTION 4 Or close the associated block valve and open its circuit breaker.

ACTION 5 With the number of OPERABLE Channels less than required by the Minimu Channels OPERABLE requirements, initiate the preplanned alternate meth of monitoring the appropriate parameter(s), within 72 hours, and:

1) eithe restore the inoperable channel(s) to OPERABLE status within 7 da of the event, or

2) prepare and submit a Special Report to the Commission pursuant Specification 6.9.3 within 30 days following the event outlining the acti taken, the cause of the inoperability, and the plans and schedule f restoring the system to OPERABLE status.

ACTION 6 With one hydrogen monitor inoperable, restore the inoperable monitor OPERABLE status within 30 days or be in at least HOT SHUTDOWN within tl next 6 hours.

ACTION 7 With both hydrogen monitors inoperable, restore at least one monitor OPERABLE status within 72 hours or be in at least HOT SHUTDOWN with the next 6 hours.

3.8 STEAM AND POWER CONVERSION SYSTEMS

Applicability: Applies to the operating status of the steam and power conversion systems.

Objective: To define conditions of the steam-relieving capacity and auxiliary feedwater system.

Specification:

1. When the reactor coolant of a nuclear unit is heated **above 350°F,** the following conditions must be met:

   a. **TWELVE (12)** of its steam generator safety valves shall be operable (except for testing).

   b. Its condensate storage tank shall contain a minimum of **185,000** gallons of water.

   c. Its main steam stop valves shall be operable and capable of closing in **5 seconds or less.**

   d. System piping, interlocks and valves directly associated with the related components in TS 3.8.1 a, b, c shall be operable.

2. The iodine-131 activity on the secondary side of a steam generator shall not exceed **0.67 µCi/gm.**

3. With the reactor coolant system above **350°F,** if any of above specifications cannot be met within **48 hours,** the reactor shall be shutdown and the reactor coolant temperature reduced below 350°F. **Specification 3.0.1 applies.**

4. The following number of independent steam generator auxiliary feedwater trains and their associated flow paths (steam and water) shall be operable when the reactor coolant is heated **above 350°F:**

a. Single Nuclear Unit Operation

Two independent auxiliary feedwater trains capable of being powered from an operable steam supply.

b. Dual Nuclear Unit Operation

Two independent auxiliary feedwater trains and a third pump capable of being powered from, and supplying water to either train.

c. If in accordance with TS 4.10.1, testing is required during start-up of either unit, TS 3.8.4.a. or b., as applicable, shall apply for an auxiliary feedwater pump, pumps, or associated flow paths (steam and water) found to be inoperable.

5. **During power operation,** if any of the conditions of 3.8.4 cannot be met, the reactor shall be shutdown and the reactor coolant temperature reduced **below 350°F,** unless one of the following conditions can be met:

a. **For single unit operation** with **one of the two** required independent auxiliary feedwater trains inoperable, restore the inoperable train to operable status within **72 hours** or the reactor shall be shutdown and the reactor coolant temperature reduced **below 350°F within the next 12 hours.**

b. **For dual unit operation, one** auxiliary feedwater pump and **its associated piping, valves, and interlocks** may be inoperable provided **two** independent auxiliary feedwater trains remain operable for time period **not to exceed 72 hours.** If the inoperable pump cannot be made operable **within 72 hours, one** reactor shall be shutdown and its reactor coolant temperature reduced **below 350°F within the next 12 hours.**

c. **For dual unit operation,** with **one** independent auxiliary feedwater train inoperable in **one** reactor, the affected reactor shall be SHUTDOWN and its reactor coolant temperature **reduced below 350°F within 72 hours.** TS 3.8.5.a applies for the single unit still in operation.

d. **For dual unit operation,** with one independent auxiliary feedwater train inoperable in both units, **one** reactor shall be SHUTDOWN and its reactor coolant temperature **reduced below 350°F within 12 hours.** TS 3.8.5.a applies for the single unit still in operation.

TABLE 4.1-1 SHEET 2

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S†	N.A.	N.A.	With analog Rod Position
11.	Steam Generator Level	S†	R	M†	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	W	R	N.A.	
15.	Refueling Water Storage Tank Level	W†	R	N.A.	
16.	Volume Control Tank Level	N.A.	R	N.A.	
17A.	Containment Pressure - Narrow Range	M††	R	N.A.	
17B.	Containment Pressure - Wide Range	M††	R	N.A.	
18A.	Process Radiation***	D	N.A.	M	
18B.	Area Radiation	D	A	M	
19.	Boric Acid Control	N.A.	N.A.	R	
20.	Containment Sump Level	N.A.	R	N.A.	
21.	Accumulator Level and Pressure	S†	R	N.A.	
22.	Steam Line Pressure	S†	R	M†	

TABLE 4.1-1 SHEET 3

	Channel Description	Check	Calibrate	Test	Remarks
23.	Logic Channels	N.A.	N.A.	M†	
24.	Emer. Portable Survey Instruments	N.A.	A	M	
25.	Seismograph	N.A.	N.A.	Q	Make trace. Test battery (change semi-annually)
26.	Auxiliary Feedwater Flow Rate	M†	R	N.A.	
27.	RCS Subcooling Margin Monitor	M†	R	N.A.	
28.	PORV Position Indicator (Primary Detector)	M†	N.A.	R	Check consists of monitoring indicated position and verifying by observation of related parameters.
29.	PORV Block Valve Position Indicator	M†	N.A.	R	
30.	Safety Valve Position Indicator	M†	R	N.A.	
31.	Loss of Voltage (both 4kv bussess)	N.A.	N.A.	R	For AFW actuation at power only
32.	Trip of both Main Feedwater Pump Breakers	N.A.	N.A.	R	For AFW actuation at power only
33.	Containment Water Level (Narrow Range)	M††	R	N.A.	
34.	Containment Water Level (Wide Range)	M††	R	N.A.	

TABLE 4.1-1 SHEET 4

	Channel Description	Check	Calibrate	Test	Remarks
35.	Containment High Range Area Radiation	S††	R(Note 1)	M††	
36.	Containment Hydrogen Monitors	S†	Q(1)	M†	(1) Channel calibration using sample gas containing: a. One volume percent hydrogen, balance nitrogen. b. Four volume percent hydrogen, balance nitrogen.
37.	High Range Noble Gas Effluent Monitors				
	a. Plant Vent Exhaust	S	R	M	
	b. Unit 3 Spent Fuel Pit Exhaust	S	R	M	
	c. Condenser Air Ejectors	S†	R	M†	
	d. Main Steam Lines	S†	R	M†	
38.	Incore Thermocouples (Core Exit Thermocouples)	M††	R	N.A.(See Note 2)	

TABLE 4.1-1 SHEET 5

* - Using moveable in-core detector system.

** - Frequency only

*** - Applies to containment particulate (R11) and gaseous (R12) monitors only. For effluent monitors, refer to Tables 4.1-3 and 4.1-4.

PR - Prior to each release

S - Each Shift

D - Daily

W - Weekly

4/M - At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year

B/W - Every Two Weeks

M - Monthly

Q - Quarterly

P - Prior to each startup if not done previous week

R - Each Refueling Shutdown

A - Annually

N.A. - Not applicable

† - N.A. during cold or refueling shutdowns. The specified tests, however, shall be performed within one surveillance interval prior to startup.

†† - N.A. during cold or refueling shutdowns. The specified tests, however, shall be performed within one surveillance interval prior to heatup above 200F.

NOTES: 1) Acceptable criteria for calibration is provided in Table II.F.1-3 of NUREG 0737.

2) Compliance will depend on instrumentation operability.

2. POST ACCIDENT CONTAINMENT VENT SYSTEM

1. Operating Tests

Operating tests shall be performed during refueling but not longer than 18 months. The tests shall consist of visual inspection of the system, operation of all valves and pressure drop and air flow measurements. Visual inspection shall include a search for any foreign materials and gasket deterioration in the HEPA filters and charcoal adsorbers. Less than 6" of water pressure drop at 55 cfm flow shall constitute acceptable performance.

2. Performance Tests

a. A visual inspection of the sytem shall be made before each DOP test and halogenated hydrocarbon leak test. At least once per 18 months or after 720 hours of system operation, in-place DOP and halogenated hydrocarbon tests at design flow (55 cfm $\pm$ 10%) and carbon analysis, or carbon replacement, for the Post Accident Containment Vent filters shall be performed. In addition, carbon analysis (or carbon replacement), DOP and halogenated hydrocarbon tests at design flow (55 cfm $\pm$ 10%) shall be performed after (1) any structural maintenance on system housings which might have affected filter bank efficiency, (2) after complete or partial replacement of a filter bank or (3) after exposure of the filters to effluents from painting, fire or chemical release. Removal of $\geq$99% DOP and $\geq$99% halogenated hydrocarbon shall constitute acceptable performance.

b. Laboratory carbon sample analysis shall show $\geq$90% methyl radio-iodine removal or the charcoal shall be replaced with charcoal that meets or exceeds the criteria of position C.6.a of Regulatory Guide 1.52 (Revision 2). The sample shall be taken in accordance with position C.6.b of Regulatory Guide

1.52. Carbon analysis shall be performed in accordance with ANSI N510-1975. Analysis shall verify the above removal efficiency for radio-iodine within 45 days after removal of the sample.

3. CONTROL ROOM VENTILATION (EMERGENCY INTERNAL CLEANUP) SYSTEM

1. A visual inspection shall be made before each in-place DOP test, halogenated hydrocarbon leak test, and airflow distribution test. The Control Room Ventilation System shall be operated monthly for at least 15 minutes to demonstrate operability. Auto initiation of the systems operations shall be checked during refueling, but not longer than 18 months. Pressure drop measurements across the filter bank shall be made annually. Less than 6" of water pressure drop at designed flow (1,000 cfm $\pm$ 10%) across the combined HEPA filter and charcoal adsorbers shall constitute acceptable performance. A visual inspection shall include a search for any foreign materials and gasket deterioration in the HEPA filters and charcoal adsorbers.

2. Performance Tests

a. A visual inspection shall be made before each in-place DOP test, halogenated hydrocarbon leak test and airflow distribution test. At least once per 18 months or after 720 hours of system operation, in-place DOP and halogenated hydrocarbon tests at design flow (1,000 cfm $\pm$ 10%) and carbon analysis shall be performed after (1) any structural maintenance on system housings, which might have affected filter bank efficiency, (2) after complete or partial replacement of a filter bank, or (3) after operational exposure of the filters to effluents from painting, fire or chemical releases. Removal of $\geq$99% DOP and $\geq$99% halogenated hydrocarbon shall constitute acceptable performance.

b. A charcoal surveillance specimen from one of the charcoal adsorbers shall be removed and analyzed for methyl radio-iodine removal capability. The results of the laboratory carbon sample analysis shall show $\geq$90% methyl radio-iodine removal efficiency. Samples shall be taken in accordance with position C.6.b of Regulatory Guide 1.52. Carbon analysis shall be performed in accordance with ANSI N510-1975. Analysis shall verify the above removal efficiency for methyl radio-iodine within 45 days after removal of the sample. Failing this, the charcoal shall be replaced with charcoal which meets or exceeds the criteria of position C.6.a of Regulatory Guide 1.52 (Revision 2).

c. System flow rate should be verified once each 18 months, following maintenance to HEPA or charcoal housings or fire or chemical release in its ventilation zone while the system is operating.

4.19 REACTOR COOLANT VENT SYSTEM

Applicability: Applies to the periodic surveillance of the reactor coolant vent system.

Objective: To verify the operability of the system.

Specification:

1. Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

   a. Verifying all manual isolation valves in each vent path are locked in the open position.

   b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

   c. Verifying flow through the reactor coolant system vent paths during COLD SHUTDOWN or REFUELING.

6.13 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,

3. Provisions for maintenance of sampling and analysis equipment.

6.14 SYSTEMS INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventative maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.15 IODINE MONITORING

The licensee shall implement 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.

6.16 BACKUP METHODS FOR DETERMINING SUBCOOLING MARGIN

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.

B3.1 BASES FOR LIMITING CONDITIONS FOR OPERATION, REACTOR COOLANT SYSTEM

1. Operational Components

The specification requires that a sufficient number of reactor coolant pumps be operating to provide coastdown core cooling in the event that a loss of flow occurs. The flow provided will keep DNBR well above the applicable design limit*. When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the reactor coolant system volume in approximately one half hour.

Each of the pressurizer safety valves is designed to relieve 283,300 lbs. per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lifting pressure would be less than the capacity of a single valve. Also, two safety valves have capacity greater than the maximum surge rate resulting from complex loss of load. (2)

The 50°F limit on maximum differential between steam generator secondary water temperature and reactor coolant temperature assures that the pressure transient caused by starting a reactor coolant pump when cold leg temperature is $\leq$275°F can be relieved by operation of one Power Operated Relief Valve (PORV). The 50°F limit includes instrument error.

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above the applicable design limit* during all normal operations and anticipated transients. In power operation with one reactor coolant loop not in operation, this specification requires that the plant be in at least Hot Shutdown within 1 hour.

* This amendment effective as of date of issuance for Unit 3 and date of Start-up, Cycle 10, for Unit 4.

In Hot Shutdown, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat, however, single failure considerations require that two loops be operable.

In Cold Shutdown, a single reactor coolant loop or RHR coolant loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two RHR loops to be operable.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The requirement that at least one residual heat removal (RHR) loop be in operation during Refueling Shutdown ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 160 F as required during Refueling Shutdown and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution stratification.

The requirement to have two RHR loops operable when there is less than 23 feet of water above the core ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function and capabilities of the reactor coolant vent system are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

2. <u>Pressure/Temperature Limits</u>

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for prevention of brittle fracture.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile streses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curves are composite curves prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100 F per hour. The cooldown limit curves are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period.

B4.19 BASES FOR REACTOR COOLANT VENT SYSTEM

The surveillance requirements of the reactor coolant vent system are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

The performance of the specified surveillances will verify the operability of the system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 110 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO.104 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

INTRODUCTION AND BACKGROUND

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements", which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications were scheduled for implementation after December 31, 1981. The staff provided guidance on the scope of Technical Specifications for all of these items in Generic Letter 83-37. Generic Letter 83-37 was issued to all Pressurized Water Reactor (PWR) licensees on November 1, 1983. In this Generic Letter, the staff requested licensees to:

1. review their facility's Technical Specifications to determine if they were consistent with the guidance provided in the Generic Letter, and

2. submit an application for a license amendment where deviations or absence of Technical Specifications were found.

By letter dated June 15, 1984, Florida Power and Light Company (the licensee) responded to Generic Letter 83-37 by submitting Technical Specifications (TS) change requests for Turkey Point Units 3 and 4. This evaluation covers the following TMI Action Plan items:

1. Reactor Coolant System Vents (II.B.1)
2. Post-Accident Sampling (II.B.3)
3. Long Term Auxiliary Feedwater System Evaluation (II.E.1.1)
4. Noble Gas Effluent Monitors (II.F.1.1)
5. Sampling and Analysis of Plant Effluents (II.F.1.2)
6. Containment High-Range Radiation Monitor (II.F.1.3)
7. Containment Pressure Monitor (II.F.1.4)
8. Containment Water Level Monitor (II.F.1.5)
9. Containment Hydrogen Monitor (II.F.1.6)
10. Instrumentation for Detection of Inadequate Core Cooling (II.F.2)

The TSs for Item III.D.3.4 - Control Room Habitability Requirements, have been completed by Amendment Nos. 83 and 77, issued on April 9, 1982.

EVALUATION

1. Reactor Coolant System Vents (II.B.1)

Our guidance for RCS vents identified the need for at least one operable vent path at the reactor vessel head and the pressurizer steam space, for Westinghouse reactors. Generic Letter 83-37 also provided limiting conditions for operation and the surveillance requirements for the RCS vents. The licensee has proposed TSs that are consistent with our guidance. We find the proposed TSs to be acceptable for the RCS vents. The T_{avg} requirement of 350°F was changed to 200°F to cover operation between cold shutdown and hot shutdown as per the guidance provided. The licensee's proposed TSs with this change are consistant with our guidance and are acceptable.

2. Post-Accident Sampling (II.B.3)

The guidance provided by Generic Letter 83-37 requested that an administrative program should be established, implemented and maintained to ensure that the licensee has the capability to obtain and analyze reactor coolant and containment atmosphere samples under accident conditions. The Post-Accident Sampling System is not required to be operable at all times. Administrative procedures are to be established for returing inoperable instruments to operable status as soon as practicable.

The licensee has provided a proposed revision to the TS which is consistent with the guidelines provided in our Generic Letter 83-37. We conclude that the licensee has an acceptable TS for the Post-Accident Sampling System.

3. Long Term Auxiliary Feedwater System Evaluation (II.E.1.1)

The guidance of NUREG-0737 provides for the improvement of reliability and performance of the Auxiliary Feedwater System (AFW). The proposed TSs contain limiting conditions for operation and surveillance requirements that are similar to other safety-related systems. We conclude that the proposed TSs are acceptable as they meet the intent of Generic Letter 83-37.

4. Noble Gas Effluent Monitors (II.F.1.1)

The licensee has supplemented the existing normal range monitors to provide noble gas monitoring in accordance with Item II.F.1.1. Proposed TSs were submitted that are consistent with the guidelines provided in our Generic Letter 83-37. We conclude that the TSs for Item II.F.1.1 are acceptable.

5. Sampling and Analysis of Plant Effluents (II.F.1.2)

The guidance provided by Generic Letter 83-37 requested that an administrative program should be established, implemented and maintained to ensure the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The licensee has proposed TSs that are consistent with our guidance. We conclude that the TSs for sampling and analysis of plant effluents are acceptable.

6. Containment High-Range Radiation Monitor (II.F.1.3)

The licensee has installed two in-containment monitors in each Turkey Point Unit that is consistent with the guidance of TMI Action Plan Item II.F.1.3. Generic Letter 83-37 provided guidance for limiting conditions of operation and surveillance requirements for these monitors. The licensee proposed TSs that are consistent with the guidance provided in our Generic Letter 83-37. We conclude that the proposed TSs for Item II.F.1.3 are acceptable.

7. Containment Pressure Monitor (II.F.1.4)

Each Turkey Point Unit has been provided with two supplementary channels for monitoring containment pressure following an accident. The licensee has proposed TSs that are consistent with the guidelines contained in Generic Letter 83-37. We conclude that the proposed TSs for containment pressure monitor are acceptable.

8. Containment Water Level Monitor (II.F.1.5)

Narrow range and wide range containment water level monitors provide the capability required by TMI Action Plan Item II.F.1.5. The TSs for both units contain limiting conditions of operation and surveillance requirements that are consistent with the guidance contained in Generic Letter 83-37. We conclude that the proposed TSs for containment water level monitors are acceptable.

9. Containment Hydrogen Monitor (II.F.1.6)

The licensee installed containment hydrogen monitors that provide the capability required by TMI Action Plan Item II.F.1.6. The proposed Technical Specifications contain appropriate limiting conditions of operation and surveillance for these monitors. We conclude that the proposed TSs are acceptable as they are consistent with the guidance contained in Generic Letter 83-37.

10. Instrumentation for Detection of Inadequate Core Cooling (II.F.2)

The NRC staff has not completed their review of TMI Item II.F.2 and will not issue the proposed TSs for this item until the review has been completed.

Environmental Consideration

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these 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 occupation radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 17, 1984

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