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JUL 6 1981

Docket Nos. 50-250
and 50-251

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

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. DPR-31 and Amendment No. 63 to Facility Operating License No. DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letters dated December 23, 1980 and March 10, 1981.

These amendments incorporate certain of the lessons learned Category A requirements into the Technical Specifications.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed By:

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 70 to DPR-31
2. Amendment No. 63 to DPR 41
3. Safety Evaluation
4. Notice of Issuance

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See ~~Next~~ Next Page

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Robert E. Uhrig
Florida Power and Light Company

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State of Florida
2600 Blair Stone Road
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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. 63
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 December 23, 1980, supplemented on March 10, 1981, 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 Appendices A and B, as revised through Amendment No. 63, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: JUL 6 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
-	3.1-1a
3.2-4	3.2-4
3.3-1	3.3-1
3.5-1	3.5-1
Table 3.5-2	Table 3.5-2
Table 3.5-4	Table 3.5-4
-	Table 3.5-5
Table 4.1-1	Table 4.1-1
Table 4.1-1 sheet 3	Table 4.1-1 sheet 3
-	Table 4.1-1 sheet 4
4.10-1	4.10-1
Table 6.2-1	Table 6.2-1
6.5	6.5
6.30	6.30

d. Presssurizer

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 350F.

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 350F.

2. If the average coolant temperature is greater than 350F 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 350F 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.

(b) (1) The measurement of total peaking factor, F_q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

(2) The measurement of the enthalpy rise hot channel factor, F^N_H, shall be increased by four percent to account for measurement error.

If the measured hot channel factor exceeds its limit specified under Item 7a, the reactor power shall be reduced so as not to exceed a fraction of the rated value equal to the ratio of the F_q or F^N_H limit to measured value, whichever is less. and the high neutron flux trip setpoint shall be reduced by the same ratio. If subsequent incore mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing. The reactor may be returned to higher power levels when measurements indicate that hot channel factors are within limits.

c. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter. If the axial flux difference has not been measured in the last effective full power month, the target flux difference must be updated monthly by linear interpolation using the most recent measured value and the value predicted for the end of the cycle life.

d. Except during physics tests or during excore calibration procedures and as modified by items 6e through 6g below, the indicated axial flux difference shall be maintained with a $\pm$ 5% band about the target flux difference (this defines the target band on axial flux difference).

e. If the indicated axial flux difference at a power level greater than 90% of rated power deviates

3.3 CONTAINMENT

Applicability: Applies to the integrity of the containment.

Objective: To define the operating status of the containment

Specification:

1. CONTAINMENT INTEGRITY

a. The containment integrity (as defined in 1.5) shall not be violated unless the reactor is in the cold shutdown condition. Specification 3.0.1 applies to 3.3.1.

b. The containment integrity shall not be violated when the reactor vessel head is removed unless the reactor is in the refueling shutdown condition.

2. INTERNAL PRESSURE

If the internal pressure exceeds 3 psig or the internal vacuum exceeds 2 psig, the condition shall be corrected within 8 hours or the reactor shall be brought to hot shutdown.

3. CONTAINMENT ISOLATION VALVES

With $K_{eff} \geq 0.99$, % thermal power excluding decay heat ≥ 0, and an average coolant temperature $T_{avg} \geq 200F°$, the following conditions shall be met:

The containment isolation valves for Phase containment isolation, Phase B containment isolation, and Containment Ventilation Isolation shall be operable with the isolation times of each power operated or automatic valve within the limits established for testing in accordance with Section XI of ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i), or the valve is closed.

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.

2. With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels shown in Table 3.5-5, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in a condition with K_{eff} <0.99, % thermal power excluding decay heat equal to zero, and an average coolant temperature T_{avg} <350°F within the next 12 hours.

3. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.5-5, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in a condition with K_{eff} <0.99, % thermal power excluding decay heat equal to zero, and an average coolant temperature T_{avg} <350°F within the next 12 hours.

TABLE 3.5-2

ENGINEERED SAFETY FEATURES ACTUATION

NO.	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 MIN. DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
1.	SAFETY INJECTION			
1.1	Manual	1	0	Cold Shutdown
1.2	High Containment Pressure	2	1	Cold Shutdown
1.3	High Differential Pressure between any Steam Line and the Steam Line Header	2	1	Cold Shutdown
1.4	Pressurizer Low Pressure*	2	1	Cold Shutdown
1.5	High Steam Flow in 2/3 Steam Lines with Low T_{avg} or Low Steam Line Pressure	1/line in each of 2 lines	1	Cold Shutdown
2.	CONTAINMENT SPRAY			
2.1	High Containment Pressure and High-High Containment Pressure (Coincident)	2 per set	1/set	Cold Shutdown
3.	AUXILIARY FEEDWATER			
3.1	Low-Low Steam Generator Level	2	1	Hot Shutdown
3.2	Loss of Voltage (both 4KV busses)	2	0	Cold Shutdown
3.3	Safety Injection	(---See 1 above---)		
3.4	Trip of both Main Feedwater Pump Breakers	2	0	Cold Shutdown

* This signal may be manually bypassed, when the reactor is shut down and pressure is below 2000 psig.

TABLE 3.5-4

ENGINEERED SAFETY FEATURE

SET POINTS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SET POINT
1.	High Containment Presure	Safety Injection Containment Spray* Steam Line Isolation* Containment Isolation*	$\leq$ 6 psig
2.	High-High Containment Pressure	See No. 1	$\leq$ 30 psig
3.	Pressurizer Low Pressure	Safety Injection	$\geq$ 1715 psig
4.	High Steam Line Differential Pressure (2/3 between any header and any line)	Safety Injection	$\leq$ 150 psi
5.	High Steam Line Flow (2/3 lines)	Safety Injection Steam Line Isolation	d/p for 3.84×10^6 lb/hr, 770 psig, 100% RP d/p for 0.64×10^6 lb/hr 1005 psig, 0% RP d/p linear with 1st stg. press., 0-100% RP
	Coincident with:		
	Low Steam Line Pressure, or		$\geq$ 600 psig
	Low T_{avg}.		$\geq$ 531 F
6.	Low-Low Steam Generator Level	Auxiliary Feedwater	$\geq$ 15% narrow range
7.	Loss of Voltage (both 4 KV busses)	Auxiliary Feedwater	N. A.
8.	Safety Injection	Auxiliary Feedwater	All SI setpoints
9.	Trip of both Main Feedwater Pump Breakers	Auxiliary Feedwater	N A.

* High and High-High coincident

TABLE 3.5-5

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Pressurizer Water Level	2	1
2.	Auxiliary Feedwater Flow Rate	2 per generator	1 per generator
3.	Reactor Coolant System Subcooling Margin Monitor	2*	1*
4.	PORV Position Indicator (Primary Detector)	1/valve	1/valve+
5.	PORV Block Valve Position Indicator	1/valve	1/valve+
6.	Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve

NOTE: Not effective until installed.

* For the purpose of this Specification, the pressure and temperature inputs to the Reactor Coolant System Subcooling Margin Monitor are redundant.

\+ Or close the associated block valve and rack out its circuit breaker.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
1.a.	Nuclear Power Range (Check, Calibrate and Test only applicable above 10% of rated power.)	S(1) M*(4)	D(2) Q*(4)	M(3)	1) Load vs. flux curve 2) Thermal power calculation 3) Signal to ΔT; bistable action (permissive, rod stop, trips) 4) Upper & lower detectors for symmetric offset (+5 to -5%).
b.	Power Distribution Map			M(1)	1) Following initial loading and prior to operation above 75% power. 2) Once per effective full power month. 3) Confirm hot channel factor limits.
2.	Nuclear Intermediate Range	S(1)†	N.A.	P(2)	1) Once/shift up to 50% RP. 2) Log level;bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S(1)	N.A.	P(2)	1) Once/shift when in service. 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S†	R	B/W(1)† (2)†	1) Overtemperature- ΔT 2) Overpower- ΔT
5.	Reactor Coolant Flow	S†	R	M†	
6.	Pressurizer Water Level	M†	R	M†	
7.	Pressurizer Pressure	S†	R	M†	
8.	4 kv Voltage & Frequency	N.A.	R**	R	Reactor protection circuits only
9.	Analog Rod Position	S†	R	M†	With step counters.

TABLE 4.1-1 SHEET 3

	Channel Description	Check	Calibrate	Test	Remarks
23.	Environmental Radiological Monitors	N.A.	A(1)	M(1)	(1) Flow
24.	Logic Channels	N.A.	N.A.	M†	
25.	Emer. Portable Survey Instruments	N.A.	A	M	
26.	Seismograph	N.A.	N.A.	Q	Make trace. Test battery (change semi-annually)
27.	Auxiliary Feedwater Flow Rate	M†	R	N.A.	
28.	RCS Subcooling Margin Monitor	M†	R	N.A.	
29.	PORV Position Indicator (Primary Detector)	M†	N.A	R	Check consists of monitoring indicated position and verifying by observation of related parameters
30.	PORV Block Valve Position Indicator	M†	N.A	R	
31.	Safety Valve Position Indicator	M†	R	N.A.	
32.	Loss of Voltage (both 4kv busses)	N.A.	N.A.	R	For AFW actuation at power only
33.	Trip of both Main Feedwater Pump Breakers	N.A.	N.A.	R	For AFW actuation at power only

TABLE 4.1-1 SHEET 4

*	Using moveable in-core detector system.
**	Frequency only
***	Effluent monitors only. Calibration shall be as specified in 3.9.

S	-	Each Shift
D	-	Daily
W	-	Weekly
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 appliable
†	-	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.

4.10 AUXILIARY FEEDWATER SYSTEM

Applicability: Applies to periodic testing requirements of the auxiliary feedwater system.+

Objective: To verify the operability of the auxiliary feedwater system and its ability to respond properly when required.

Specifications:

1. Each turbine-driven auxiliary feedwater pump shall be started at intervals not greater than one month; run for 15 minutes and a flow rate of 600 gpm established to the steam generators.
2. The auxiliary feedwater discharge valves shall be tested by operator action during pump tests.
3. Steam supply and turbine pressure valves shall be tested during pump tests.
4. These tests shall be considered satisfactory if control panel indication and visual observation of equipment demonstrate that all components have operated properly.
5. At least once per 18 months:
   a. Verify that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.
   b. Verify that each auxiliary feedwater pump receives a start signal as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

+ N.A. during cold or refueling shutdowns (only for the Unit at cold or refueling shutdown). The specified tests, however, shall be performed within one surveillance interval prior to starting the turbine.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY QUALIFICATIONS	One or Two Units Operating[A]	All Units Shutdown
SRO*	2	1**
RO	3	2
Non-Licensed Auxiliary Operators	3	3
Shift Technical Advisor	1+	None Required

\+ This position may be filled by one of the SRO's above, provided the individual meets the qualification requirements of 6.3.1

* Includes the licensed Senior Reactor Operator serving as Shift Supervisor.

** Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising the movement of any component within the reactor presssure vessel with the vessel head removed and fuel in the vessel.

A Operating is defined as $K_{eff} \geq 0.99$, % thermal power excluding decay heat greater than or equal to zero, and an average coolant temperature $T_{avg} \geq 200°F$.

\# Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. ALL CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. At least three (3) persons shall be maintained on site at all times for Fire Emergency response. This excludes two (2) members of the shift crew.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions. except for the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents.

6.3.2 HEALTH PHYSICS SUPERVISOR QUALIFICATIONS

6.3.2.1. The Health Physics Supervisor at the time of appointment to the position, shall, except as indicated below, meet the following:

1. He shall have a bachelor's degree or equivalent in a science or engineering subject, including some formal training in radiation protection.
2. He shall have five years of professional experience in applied radiation protection, where a master's degree in a related field is equivalent to one year experience and a doctor's degree in a related field is equivalent to two years of experience.
3. Of his five years of experience, three years shall be in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered at Turkey Point Plant.

6.3.2.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines, and the NRC Office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.2.1.

6.4 TRAINING

6.4.1 A retaining and replacement program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5, ANSI N18.1-1971 and Appendix A to 10 CFR Part 55.

6.13 ENVIRONMENTAL QUALIFICATIONS

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines): or, NUREG-0588 "Interim Staff Position on Environmental Qualifcation of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to the Order of Modification of License Nos. DPR-31 and DPR-41 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification methods used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-31
AND AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-41
FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT PLANT UNIT NOS. 3 AND 4
DOCKET NOS. 50-250 AND 50-251

I. INTRODUCTION

By letters dated December 23, 1980, and supplemented on March 10, 1981, Florida Power and Light Company (the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License Nos. DPR-31 and DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in direct response to the NRC staff's letter dated July 2, 1980.

II. BACKGROUND INFORMATION

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter dated April 7, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TS to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

III. EVALUATION

2.1.1 Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The licensee has (has provided) the requisite emergency power supplies. The licensee has proposed adequate TSs which provide for a 31-day channel check and 18-month channel calibration and actions in the event of component inoperability. We have reviewed these proposed TSs and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components and are thus acceptable.

2.1.3.a Direct Indication of (of Flow) Valve Position

The licensee has provided a direct indication of power-operated relief valve (PORV) and safety valve position in the control room. These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs with a 31-day channel check and an 18-month channel calibration requirement; thus, the TSs are acceptable and they meet our July 2, 1980 model TS criteria.

2.1.3.b Instrumentation for Inadequate Core Cooling

The licensee has installed an instrument system to detect the effects of low reactor coolant level and inadequate core cooling. These instruments, sub-cooling meters, receive and process data from existing plant instrumentation. We previously reviewed this system in our Safety Evaluation dated April 7, 1980. The licensee submitted TSs with a 31-day channel check and an 18-month channel calibration requirement and actions to be taken in the event of component inoperability. We conclude the TSs are acceptable as they meet our July 2, 1980 model TS criteria.

2.1.4 Diverse Containment Isolation

The licensee has modified the containment isolation system so that diverse parameters will be sensed to ensure automatic isolation of non-essential systems under postulated accident conditions. These parameters are safety inspection or main steam isolation. We have reviewed this system in our Lessons Learned Category "A" Safety Evaluation dated April 7, 1980. The modification is such that it does not result in the automatic loss of containment isolation after the containment isolation signal is reset. Reopening of containment isolation would require deliberate operator action.

2.1.7a <u>Auto Initiation of Auxiliary Feedwater Systems</u>

The plant has provision for the automatic initiation of auxiliary (emergency) feedwater flow on loss of normal feedwater flow. The TSs submitted by the licensee list the appropriate components, describe the tests and provide for proper test frequency. The TSs contain appropriate actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

2.1.7.b <u>Auxiliary (Emergency) Feedwater Flow Indication</u>

The licensee has installed auxiliary (emergency) feedwater flow indication that meets our testability and vital power requirements. We reviewed this system in our Safety Evaluation dated April 7, 1980. The licensee has proposed a TS with 31-day channel check and 18-month channel calibration requirements. We find this TS acceptable as it meets the criteria of our July 2, 1980 model TS criteria.

2.2.1.b <u>Shift Technical Advisor (STA)</u>

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The licensee's application would add one STA to each shift to perform the function of accident assessment. The individual performing this function will have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. Part of the STA duties are related to operating experience review function. Based on our review, we find the licensee's submittal to satisfy our requirements and is acceptable.

<u>EVALUATION TO SUPPORT ADMINISTRATIVE CONDITIONS</u>

2.1.4 <u>Integrity of Systems Outside Containment</u>

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment. By letter dated March 10, 1981, the licensee agreed to adopt such an administrative condition; accordingly we have included this condition in the TSs.

2.1.8.c <u>Iodine Monitoring</u>

Our letter dated July 2, 1980, indicated that the license should be amended by adding a condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditons. By letter dated March 10, 1981, the licensee agreed to adopt such an administrative condition; accordingly, we have included this condition in the TSs.

2.1.3.b Backup Method for Determining Subxooling Margi.

Our letter of July 2, 1980, indicated that the license should be amended by adding a condition related to the determination of subcooling margin; this is a precursor to warn of inadequate core cooling in the event of an accident. Such a condition would require the training of personnel and the generation of procedures to accurately monitor the reactor coolant system subcooling margin. By letter dated March 10, 1981, the licensee agreed to adopt such an administrative condition; accordingly, we have included this condition in the TSs.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: JUL 6 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-250 AND 50-251

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 70 to Facility Operating License No. DPR-31, and Amendment No. 63 to Facility Operating License No. DPR-41 issued to Florida Power and Light Company (the licensee), which revised Technical Specifications for operation of Turkey Point Plant, Unit Nos. 3 and 4 (the facilities) located in Dade County, Florida. The amendments are effective as of the date of issuance.

The amendments incorporate certain of the lessons learned Category A requirements into the Technical Specifications.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated December 23, 1980, as supplemented March 10, 1981, (2) Amendment Nos. 70 and 63 to License Nos. DPR-31 and DPR-41, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Environmental and Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 6th day of July, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operationg Reactors Branch No. 1
Division of Licensing