January 11, 1978

Docket No.: 50-302

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Florida Power Corporation
ATTN: Mr. W. P. Stewart, Director
Power Production
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Gentlemen:

The Commission has issued the enclosed Amendment No. 11 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your application dated July 15, 1977.

This amendment resolves conflicting requirements with respect to the Reactor Building Purge Exhaust Duct Monitor trip setpoints, amplifies an emergency feedwater pump surveillance requirement, corrects typographical errors and deletes improperly characterized valves from a table of containment isolation valves. Those portions of your proposal dealing with deletion of sodium thiosulfate tank requirements and changes to Section 6 of your Technical Specifications are being handled separately.

Changes to your proposal were necessary to meet our requirements. These have been discussed with and agreed to by your staff.

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

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 11
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

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DATE ➤	1/3/78	1/4/78	1/9/78	1/6/78	1/10/78	1/11/78

Florida Power Corporation

cc w/enclosures:

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Vice President and General Counsel
P. O. Box 14042
St. Petersburg, Florida 33733

Mr. Wilbur Langely, Chairman
Board of County Commissioners
Citrus County
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Chief, Energy Systems
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U. S. Environmental Protection Agency
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Crystal River Public Library
Crystal River, Florida 32629

cc w/enclosures and incoming
dtd.: 7/15/77
Bureau of Intergovernmental Relations
660 Apalchee Parkway
Tallahassee, Florida 32304

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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al (the licensees) dated July 15, 1977, 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 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 11, are hereby incorporated in the license. Florida Power Corporation 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

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 11, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 11

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 3-12

3/4 3-16

3/4 3-20

3/4 3-23

3/4 3-25

3/4 6-19

3/4 7-4

3/4 7-5

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. RCS Pressure Low-Low	3	2	2	1, 2, 3**	9#
4. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
2. REACTOR BLDG. COOLING AND ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	13
b. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. REACTOR BLDG. SPRAY					
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	3	2	2	1, 2, 3	12
b. Automatic Actuation Logic	2	1	2	1, 2, 3	10
4. OTHER SAFETY SYSTEMS					
a. Reactor Bldg. Purge Exhaust Duct Isolation on High Radioactivity					
Gaseous	1	1	1	1, 2, 3, 4	11#

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION		
a. High Pressure Injection ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
3. RCS Pressure Low	$\geq$ 1500 psig	$\geq$ 1500 psig
4. RCS Pressure Low-Low	$\geq$ 500 psig	$\geq$ 500 psig
5. Automatic Actuation Logic	Not Applicable	Not Applicable
b. Low Pressure Injection ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
3. RCS Pressure Low-Low	$\geq$ 500 psig	$\geq$ 500 psig
4. Automatic Actuation Logic	Not Applicable	Not Applicable
2. REACTOR BLDG. COOLING & ISOLATION		
a. ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
3. Automatic Actuation Logic	Not Applicable	Not Applicable
b. ES Actuation Indication "AB"		
1. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. REACTOR BLDG. SPRAY		
a. Reactor Bldg. Pressure High-High	≤ 30 psig	≤ 30 psig
coincident with HPI Signal	See 1.a.2, 3, 4	See 1.a.2, 3, 4
b. Automatic Actuation Logic	Not Applicable	Not Applicable
4. OTHER SAFETY SYSTEMS		
a. Reactor Bldg. Purge Exhaust Duct Isolation on High Radioactivity		
Gaseous	*	Not Applicable
b. Steam Line Rupture Matrix		
1. Low SG Pressure	≥ 600 psig	≥ 600 psig
2. Automatic Actuation Logic	Not Applicable	Not Applicable

*Determined by requirements of Appendix "B" Tech. Specs. Section 2.4.2 - Crystal River 3 Operating License No. DPR-72.

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. RCS Pressure Low-Low	S	R	M	1, 2, 3
4. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3, 4
2. REACTOR BLDG. COOLING AND ISOLATION				
a. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4
b. Reactor Bldg. Pressure High	S	R	M(2)	1, 2, 3
c. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3, 4

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. REACTOR BLDG. SPRAY				
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	S	R	M(4)	1, 2, 3
b. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3
4. OTHER SAFETY SYSTEMS				
a. Reactor Bldg. Purge Exhaust Duct Isolation on High Radioactivity				
1. Gaseous	S	Q	M	All Modes
b. Steam Line Rupture Matrix				
1. Low SG Pressure	N/A	R	N/A	1, 2, 3
2. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area					
i. Criticality Monitor	1	*	$\leq$ 15 mR/hr	$10^{-1} - 10^{4}$ mR/hr	14
2. PROCESS MONITORS					
a. Fuel Storage Pool Area					
i. Gaseous Activity - Ventilation System Isolation	1	**	$\leq$ 2 x background	$10^{1} - 10^{6}$ cpm	16
b. Reactor Building					
i. Gaseous Activity -					
a) Purge Exhaust Duct Isolation	1	6	***	$10^{1} - 10^{6}$ cpm	17
b) RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10^{1} - 10^{6}$ cpm	15
ii. Iodine Activity -					
RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10^{1} - 10^{6}$ cpm	15

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool
*** Determined by requirements of Appendix "B" Tech. Specs. Section 2.4.2 - Crystal River 3 Operating License NO. DPR-72.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area				
i. Criticality Monitor	S	R	M	*
2. PROCESS MONITORS				
a. Fuel Storage Pool Area				
i. Gaseous Activity - Ventilation System Isolation	S	R	M	**
b. Reactor Building				
i. Gaseous Activity -				
a) Purge Exhaust Duct Isolation	S	Q	M	6
b) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Iodine Activity - RCS Leakage Detection	S	R	M	1, 2, 3, & 4

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 As a minimum, the incore detectors shall be OPERABLE as specified below.

a. For AXIAL POWER IMBALANCE measurements:

   1. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane with one plane at the core mid-plane and one plane in each axial core half.

   2. The axial planes in each core half shall be symmetrical about the core mid-plane.

   3. The detector strings shall not have radial symmetry.

b. For QUADRANT POWER TILT measurements with the Minimum Incore Detector System:

   1. Two sets of 4 detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

   2. Detectors in the same plane shall have quarter core radial symmetry.

c. For QUADRANT POWER TILT measurements with the Symmetric Incore Detector System at least 75% of the detectors in each core quadrant shall be OPERABLE.

APPLICABILITY: When the incore detection system is used for surveillance of:

a. The AXIAL POWER IMBALANCE, or

b. The QUADRANT POWER TILT.

ACTION:

With less than the specified minimum incore detector arrangement OPERABLE, do not use incore detector measurements to determine AXIAL POWER IMBALANCE or QUADRANT POWER TILT. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME (seconds)
9.	(Continued)		
	MUV-163 check #	open during HPI and	NA
	MUV-25 #	iso. dur. nor. operation	60
	MUV-164 check #		NA
	MUV-26 #		60
	MUV-160 check #	open during HPI and	NA
	MUV-23 #	iso. dur. nor. operation	60
	MUV-161 check #	open during HPI and	NA
	MUV-24 #	iso. dur. nor. operation	60
	MUV-27 #	open dur. nor. operation and closed during HPI	60
10.	SWV-39 #	iso. NSCCC from AHF-1C	60
	SWV-45 #		60
	SWV-35 #	iso. NSCCC from AHF-1A	60
	SWV-41 #		60
	SWV-37 #	iso. NSCCC from AHF-1B	60
	SWV-43 #		60
	SWV-48 #	to isolate NSCCC from	60
	SWV-47 #	MUHE-1A & 1B and WDT-5	60
	SWV-49 #		60
	SWV-50 #		60
	SWV-80 #	iso. NSCCC from RCP-1A	60
	SWV-84 #		60
	SWV-82 #	iso. NSCCC from RCP-1C	60
	SWV-86 #		60
	SWV-81 #	iso. NSCCC from RCP-1D	60
	SWV-85 #		60
	SWV-79 #	iso. NSCCC from RCP-1B	60
	SWV-83 #		60
	SWV-109#	NSCCC to DRRD-1	60
	SWV-110#		60

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER			FUNCTION	ISOLATION TIME (seconds)
	11.	WDV-4	iso. WDT-4 from RB sump	60
		WDV-3		60
		WDV-60 & 61	iso. WDT-4 from WDT-5	60
		WDV-94 & 62	iso. WDT-4 from WDP-8	60
		WDV-406	iso. gas waste disposal	60
		WDV-405	from vents in RC system	60
	12.	WSV-3	iso. containment monitoring	60
		WSV-4	system from RB	60
		WSV-5		60
		WSV-6		60
B.	CONTAINMENT PURGE AND EXHAUST			
	1.	AHV-1C & 1D	iso. pur. sup. system	60
		AHV-1B & 1A	iso. pur. exhaust system	60
C.	MANUAL			
	1.	IAV-28	iso. IA from RB	NA
		IAV-29		NA
	2.	LRV-50	iso. leak rate test system	NA
		LRV-36	from RB	NA
		LRV-51	iso. atmos. vent and RB	NA
		LRV-35 & 47	purge exhaust system from RB	NA
		LRV-49	iso. atmos. vent from RB	NA
		LRV-38 & 52		NA
		LRV-45	iso. LR test panel from RB	NA
		LRV-44		NA
	3.	MSV-146#	iso. misc. waste storage tank from RCSG-1B	NA
	4.	NGV-62	iso. NG system from	NA
		NGV-81 #	steam generators	NA
		NGV-82	iso. NG system from pzr.	NA

TABLE 4.7-1

STEAM LINE SAFETY VALVES

VALVE NUMBER	LIFT SETTING (± 1%) (psig)	ORIFICE SIZE (inches)
STEAM GENERATOR 3A		
Main steam line A1		
MSV - 34	1050	4.515
MSV - 38	1070	4.515
MSV - 43	1090	4.515
MSV - 40	1100	3.750
Main steam line A2		
MSV - 33	1050	4.515
MSV - 37	1070	4.515
MSV - 42	1090	4.515
MSV - 46	1100	4.515
STEAM GENERATOR 3B		
Main steam line B1		
MSV - 35	1050	4.515
MSV - 39	1070	4.515
MSV - 44	1090	4.515
MSV - 47	1100	4.515
Main steam line B2		
MSV - 36	1050	4.515
MSV - 41	1070	4.515
MSV - 45	1090	4.515
MSV - 48	1100	3.750

PLANT SYSTEMS

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent steam generator emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. One emergency feedwater pump capable of being powered from an OPERABLE emergency bus, and

b. One emergency feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one emergency feedwater system inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the steam turbine driven pump develops a discharge pressure of $\geq$ 1100 psig on recirculation flow when the secondary steam supply pressure is greater than 200 psig.*

*When not in MODES 1, 2, or 3, surveillance shall be performed within 24 hours after entering MODE 3 and prior to entering MODE 2.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on an emergency feedwater actuation test signal.

2. Verifying that the steam turbine driven pump starts automatically upon receipt of an emergency feedwater actuation test signal.

3. Verifying that the operating air accumulators for FWV-39 and FWV-40 maintain $\geq$ 27 psig for at least one hour when isolated from their air supply.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum contained volume of 150,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the condenser hotwell as a backup supply to the emergency feedwater system and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume to be within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The condenser hotwell shall be demonstrated OPERABLE at least once per 12 hours by verifying a minimum contained volume of 150,000 gallons of water whenever the condenser hotwell is the supply source for the emergency feedwater system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

Introduction

By letter dated July 15, 1977, Florida Power Corporation (FPC) proposed changes to the Crystal River Unit No. 3 Technical Specifications. This proposal included changes dealing with reactor building purge exhaust duct isolation trip setpoints, emergency feedwater pump surveillance and the listing of containment isolation valves. We have evaluated the proposed changes.

Evaluation

1. FPC proposed to change the "Trip Setpoint" for Reactor Building Purge Exhaust Duct Isolation from 1 X 10^2 μCi/sec in Table 3.3-4 and $\leq$2 X background in Table 3.3-6 to "Determined by requirements of Appendix B, Section 2.4.2 - Crystal River 3 Operating License No. DPR-72." In addition, they proposed to change the required "Channel Calibration" frequency from every 18 months to quarterly. FPC states that this change will remove inconsistencies within the Appendix A Technical Specifications and between Appendix A and Appendix B Technical Specifications.

   The monitoring instrumentation referred to in Tables 3.3-3, 3.3-4, and 4.3-2 as the "Reactor Building Purge Isolation ..." is the "Containment Purge and Exhaust Isolation" in Tables 3.3-6 and 4.3-3. Both titles refer to the Reactor Building Purge Exhaust Duct Monitor's function (FSAR Section 11.4.2.1.2.a). To avoid confusion, the titles in the above tables should all read "Reactor Building Purge Exhaust Duct Isolation." FPC has agreed to this change.

   The existing setpoints of 1 X 10^2 μCi/sec and $\leq$2 X background were based on anticipated flow rates in the purge exhaust duct and expected background levels (FSAR page 11-16). These may be different from the setpoint as determined by the requirements of Appendix B, Section 2.4.2, which is based on an isotopic analysis of each release. Because compliance with Section 2.4.2 assures compliance with 10 CFR Part 20 and 10 CFR §50.36a, the trip setpoints specified in Tables 3.3-4 and 3.3-6 should be determined using Section 2.4.2. Furthermore, this change will eliminate the conflict between the three setpoints involved. Based on the above, we have determined that the change to indicate all affected setpoints are determined using Section 2.4.2 is acceptable.

Section 2.4.2 requires quarterly calibration of the Reactor Building Purge Exhaust Duct Monitor while Appendix A requires calibration every 18 months. The proposed change to Appendix A to require quarterly calibration eliminates this conflict with no decrease in the frequency of channel calibration and is therefore acceptable.

2. Currently, Technical Specification 4.7.1.2.a requires verification every 31 days that each steam turbine driven emergency feedwater pump develops a discharge pressure of ≥1100 psig on recirculation flow when the secondary steam supply pressure is greater than 200 psig. This requirement is applicable in Modes 1, 2, and 3. Prior to entry into Mode 3 while in Mode 4 (Hot Shutdown - average coolant temperature 200°F to 280°F), there is not adequate steam via the Main Steam System to run the turbine driven pump for this surveillance, as may be required by Technical Specification 4.0.4. FPC has proposed to add a footnote stating that when the plant is not in Modes 1, 2, or 3, surveillance shall be performed within 24 hours after entering Mode 3 and prior to entering Mode 2. It is the intent of this surveillance requirement to check the operability of the turbine driven pump when secondary steam supply pressure is greater than 200 psig and it is not practicable if the operational mode prohibits this initial condition. Therefore, we find this change acceptable.

   There are typographical errors in Technical Specification 3.7.1.2 which FPC has proposed to correct. These changes would correctly indicate that there is only one steam turbine driven emergency feedwater pump and that this is the only emergency feedwater pump which receives an automatic start signal. This is as stated in Chapter 10 of the FSAR and therefore correction of the typographical errors is acceptable.

3. Table 3.6-1 of the Technical Specifications, "Containment Isolation Valves," currently lists MUV-18 and associated check valve MUV-162 as containment isolation valves required to isolate the Makeup System from the Reactor Coolant Pump seals. FPC has stated and we agree that these valves are open during normal operation and high pressure injection and ~~do~~ not have an automatic isolating function via either a containment isolation signal or a containment radiation - high signal.

   The containment isolation valves required to perform a safety related function are those listed in Table 3.6-1. Technical Specification 3.6.3.1, which refers to this list, requires that these valves be operable and imposes surveillance requirements to ensure that these operate upon receipt of isolation and radiation - high signals. Since MUV-18 and 162 are not required to perform a safety related function (they may be open during high pressure injection), and do not receive containment isolation or radiation high signals (FSAR Table 5-4), removal of these valves from Table 3.6-1 is acceptable.

Environmental Consideration

We have determined that the amendment does 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 amendment involves 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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 11, 1978

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-302

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 11 to Facility Operating License No. DPR-72, issued to the Florida Power Corporation, City of Alachua, City of Bushnell, City of Gainesville, City of Kissimmee, City of Leesburg, City of New Smyrna Beach and Utilities Commission, City of New Smyrna Beach, City of Ocala, Orlando Utilities Commission and City of Orlando, Sebring Utilities Commission, Seminole Electric Cooperative, Inc., and the City of Tallahassee (the licensees) which revised Technical Specifications for operation of the Crystal River Unit No. 3 Nuclear Generating Plant located in Citrus County, Florida. The amendment is effective as of the date of issuance.

This amendment resolves conflicting requirements with respect to the Reactor Building Purge Exhaust Duct Monitor trip setpoints, amplifies an emergency feedwater pump surveillance requirement, corrects typographical errors and deletes improperly characterized valves from a table of containment isolation valves.

The application for the amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 15, 1977, (2) Amendment No. 11 to License No. DPR-72, 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 Crystal River Public Library, Crystal River, Florida. 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 Operating Reactors.

Dated at Bethesda, Maryland, this 11th day of January 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors