July 16, 1985

DMB 016

Docket No. 50-302

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Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
Post Office Box 14042; M.A.C. H-2
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

The Commission has issued the enclosed Amendment No. 78 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 23, 1985, as supplemented by letters dated June 6, 1985 and June 28, 1985.

The amendment changes the TSs for CR-3 to include requirements for the upgraded Emergency Feedwater System (EFW). The new specifications provide operability and surveillance requirements for EFW manual initiation and automatic actuation logic and are in conformance with the B&W Standard Technical Specifications.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

"ORIGINAL SIGNED BY"

Harley Silver, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 78 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

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Florida Power Corporation

Crystal River Unit No. 3 Nuclear
Generating Plant

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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. 78
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 January 23, 1985, as supplemented June 6, and June 28, 1985, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 78 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 16, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 78

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 3-14

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3/4 3-17

3/4 3-17a

3/4 3-19

3/4 3-21

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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#

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Main Feedwater and Main Steam Isolation					
1. Manual Initiation					
a. SGA MFW Isolation	4	2####	3	1,2,3	14#
b. SGB MFW Isolation	4	2####	3	1,2,3	14#
c. SGA MSL Isolation	4	2####	3	1,2,3	14#
d. SGB MSL Isolation	4	2####	3	1,2,3	14#
2. OTSG A or B Pressure Low	4/Steam Generator	2/Steam Generator	3/Steam Generator	1,2,3***	14#
3. Automatic Actuation Logic					
a. SGA MFW Isolation	2	1	2	1,2,3	10
b. SGB MFW Isolation	2	1	2	1,2,3	10
c. SGA MSL Isolation	2	1	2	1,2,3	10
d. SGB MSL Isolation	2	1	2	1,2,3	10
c. Emergency Feedwater					
1. Manual Initiation	4	2####	3	1,2,3	14#
2. MFW Pump Turbines A and B Control Oil Low	4	2	3	1##	14#
3. OTSG A or B Level Low-Low	4/Steam Generator	2/Steam Generator	3/Steam Generator	1,2,3	14#
4. OTSG A or B Pressure Low	4/Steam Generator	2/Steam Generator	3/Steam Generator	1,2,3***	14#
5. All Reactor Coolant Pumps Tripped	4	2	3	1,2####	14#
6. Automatic Actuation Logic	2	1	2	1,2,3	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
5. REACTOR BLDG. 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
d. Manual Initiation (HPI Isolation)	2	1	2	1, 2, 3, 4	13
e. RCS Pressure Low (HPI Isolation)	3	2	2	1, 2, 3*	13
f. Automatic Actuation Logic (HPI Isolation)	2	1	2	1, 2, 3, 4	10

TABLE 3.3-3 (Continued)

TABLE NOTATION

*Trip function may be bypassed in this MODE with RCS pressure below 1700 psig. Bypass shall be automatically removed when RCS pressure exceeds 1700 psig.

**Trip function may be bypassed in this MODE with RCS pressure below 900 psig. Bypass shall be automatically removed when RCS pressure exceeds 900 psig.

***Trip function may be bypassed in this MODE with steam generator pressure below 750 psig. Bypass shall be automatically removed when steam generator pressure exceeds 750 psig.

#The provisions of Specification 3.0.4 are not applicable.

##Trip automatically bypassed below 20 percent of RATED THERMAL POWER.

###Trip function may be bypassed below 10 percent of RATED THERMAL POWER.

####Manual trip function occurs if two channels in the same train are actuated.

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours; however, one channel may be bypassed for up to 1 hour for Surveillance testing per Specification 4.3.2.1.1.

ACTION 11 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the minimum channels OPERABLE required is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for Surveillance testing per Specification 4.3.2.1.1.

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour; one additional channel may be bypassed for up to 2 hours for Surveillance testing per Specification 4.3.2.1.1.

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		
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 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. REACTOR BUILDING SPRAY		
a. Reactor Building Pressure High-High coincident with HPI Signal	≤ 30 psig See 1.a.2, 3, 4	≤ 30 psig See 1.a.2, 3, 4
b. Automatic Actuation (Logic)	Not Applicable	Not Applicable
4. OTHER SAFETY SYSTEMS		
a. Reactor Building Purge Exhaust Duct Isolation on High Radioactivity		
Gaseous	*	Not Applicable
b. Main Feedwater and Main Steam Isolation		
1. Manual	Not Applicable	Not Applicable
2. OTSG A or B Pressure Low	≥600 psig	≥600 psig
3. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Emergency Feedwater		
1. Manual Initiation	Not Applicable	Not Applicable
2. MFW Pump Turbines A and B Control Oil	≥ 55 psig	≥ 55 psig
3. OTSG A or B Level Low-Low	≥ 0 inches	≥ 0 inches
4. OTSG A or B Pressure Low	≥ 600 psig	≥ 600 psig
5. All Reactor Coolant Pumps Tripped	Loss of 4 pumps (per Table 2.2-1, #8)	Loss of 4 pumps
6. Automatic Actuation Logic	Not Applicable	Not Applicable

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

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. REACTOR BLDG. ISOLATION		
a. ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	≤ 4 psig	≤ 4 psig
3. Automatic Actuation Logic	Not Applicable	Not Applicable
4. Manual Initiation (HPI Isolation)	Not Applicable	Not Applicable
5. RCS Pressure Low (HPI Isolation)	< 1500 psig	< 1500 psig
6. Automatic Actuation Logic (HPI Isolation)	Not Applicable	Not Applicable

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
1. Manual	
a. High Pressure Injection	Not Applicable
b. Low Pressure Injection	Not Applicable
c. Reactor Building Cooling	Not Applicable
d. Reactor Building Isolation	Not Applicable
e. Reactor Building Spray	Not Applicable
f. Reactor Building Purge Isolation	Not Applicable
g. MFW and MSL Isolation	
1. Emergency Feedwater Actuation	Not Applicable
2. Feedwater Isolation	Not Applicable
3. Steam Line Isolation	Not Applicable
h. Emergency Feedwater Actuation	Not Applicable
2. Reactor Building Pressure-High	
a. High Pressure Injection	25*
b. Low Pressure Injection	25*
c. Reactor Building Cooling	25*
d. Reactor Building Isolation	60*
3. Reactor Building Pressure High-High (with HPI signal)	
a. Reactor Building Spray	56*
4. RCS Pressure Low	
a. High Pressure Injection	25*
b. HPI Isolation	60*
5. RCS Pressure Low-Low	
a. High Pressure Injection	25*
b. Low Pressure Injection	25*
6. Low Steam Generator Pressure	
a. Feedwater Isolation	34
b. Steam Line Isolation	5
c. Emergency Feedwater Actuation	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
7. Containment Radioactivity-High	
a. Reactor Building Purge Isolation	15*
8. Main Feedwater Pump Turbines A and B Control Oil Low	
a. Emergency Feedwater Actuation	Not Applicable
9. OTSG A or B Level Low-Low	
a. Emergency Feedwater Actuation	50*
10. All Reactor Coolant Pumps Tripped	
a. Emergency Feedwater Actuation	Not Applicable

*Diesel Generator starting and sequence loading delays included.
Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION				
a. High Pressure Injection				
1. Manual Initiation	N/A	N/A	R	5 or 6
2. Reactor Building Pressure High	S	R	M(2)	1, 2, 3
3. RCS Pressure Low	S	R	M	1, 2, 3
4. RCS Pressure Low-Low	S	R	M	1, 2, 3
5. Automatic Actuation Logic	N/A	N/A	M(1) (3) (5)	1, 2, 3, 4
b. Low Pressure Injection				
1. Manual Initiation	N/A	N/A	R	5 or 6
2. Reactor Building Pressure High	S	R	M(2)	1, 2, 3
3. RCS Pressure Low-Low	S	R	M	1, 2, 3
4. Automatic Actuation Logic	N/A	N/A	M(1) (3)	1, 2, 3, 4
2. REACTOR BUILDING COOLING				
a. Manual Initiation	N/A	N/A	R	5or 6
b. Reactor Building Pressure High	S	R	M(2)	1, 2, 3
c. Automatic Actuation Logic	N/A	N/A	M(1) (3) (5)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. REACTOR BUILDING SPRAY				
a. Reactor Building Pressure High-High coincident with HPI Signal	S	R	M(4)	1,2,3
b. Automatic Actuation Logic	N/A	N/A	M(1)(3)(5)	1,2,3
4. OTHER SAFETY SYSTEMS				
a. Reactor Building Purge Exhaust Duct Isolation on High Radioactivity				
1. Gaseous	S	Q	M	All Modes
b. Main Feedwater and Main Steam Isolation				
1. Manual Initiation				
a. SGA MFW Isolation	N/A	N/A	M	1,2,3
b. SGB MFW Isolation	N/A	N/A	M	1,2,3
c. SGA MSL Isolation	N/A	N/A	M	1,2,3
d. SGB MSL Isolation	N/A	N/A	M	1,2,3
2. OTSG A or B Pressure Low	S	R	M	1,2,3
3. Automatic Actuation Logic				
a. SGA MFW Isolation	N/A	N/A	M(6)	1,2,3
b. SGB MFW Isolation	N/A	N/A	M(6)	1,2,3
c. SGA MSL Isolation	N/A	N/A	M(6)	1,2,3
d. SGB MSL Isolation	N/A	N/A	M(6)	1,2,3
c. Emergency Feedwater				
1. Manual Initiation	N/A	N/A	M	1,2,3
2. MFW Pump Turbine A and B Control Oil Low	S	R	M	1
3. OTSG A or B Level Low-Low	S	R	M	1,2,3
4. OTSG A or B Pressure Low	S	R	M	1,2,3
5. All Reactor Coolant Pumps Tripped	S	R	M	1,2
6. Automatic Actuation Logic	N/A	N/A	M	1,2,3

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. REACTOR BUILDING ISOLATION				
a. Manual Initiation	N/A	N/A	R	5 or 6
b. Reactor Building Pressure High	S	R	M(2)	1, 2, 3
c. Automatic Actuation Logic	N/A	N/A	M(1)(3)(5)	1, 2, 3, 4
d. Manual Initiation (HPI Isolation)	N/A	N/A	R	5 or 6
e. RCS Pressure Low (HPI Isolation)	S	R	M	1, 2, 3
f. Automatic Actuation Logic (HPI Isolation)	N/A	N/A	M(1)(3)(5)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST of the Automatic Actuation Logic need only demonstrate one combination of the three two-out-of-three logic combinations that are operable provided that a different combination is tested at each test interval, such that all three combinations will be confirmed to be operable by the time the third successive test is completed.

(2) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying pressure to the appropriate side of the transmitter.

(3) Each logic channel shall be tested at least once every other 31-day period (applies only to Test Groups HPI-3, LPI-1 and LPI-2 for the duration of Fuel Cycle 5 - see (5) below).

(4) Reactor building pressure High-High signal only.

(5) Monthly CHANNEL FUNCTIONAL TEST of the Automatic Actuation Logic circuitry has been waived for all Test Groups with the exception of Test Groups HPI-3, LPI-1 and LPI-2 for the duration of Fuel Cycle 5 for Crystal River Unit 3.

(6) The Channel functional test need not include actuation of the end device.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1.	Power Range Nuclear Flux	0-125%	2
2.	Reactor Building Pressure	0-70 psia	2
3.	Source Range Nuclear Flux	10^{-1} to 10^{6} cps	2
4.	Reactor Coolant Outlet Temperature	520° F - 620° F	2 per loop
5.	Reactor Coolant Total Flow	0-160 x 10^{6} lb./hr.	1
6.	RC Loop Pressure	0-2500 psig 0- 600 psig 1700-2500 psig	2 1 2
7.	Pressurizer Level	0-320 inches	2
8.	Steam Generator Outlet Pressure	0-1200 psig	2/steam generator
9.	Steam Generator Operating Range Level	0-100%	2/steam generator
10.	Borated Water Storage Tank Level	0-50 feet	2
11.	Startup Feedwater Flow	0-1.5x10^{6} lb./hr.	2
12.	Reactor Coolant System Subcooling Margin Monitor	-658°F to + 668°F	1
13.	PORV Position Indicator (Primary Detector)	N/A	1
14.	PORV Position Indicator (Backup Detector)	N/A	N/A
15.	PORV Block Valve Position Indicator	N/A	N/A
16.	Safety Valve Position Indicator (Primary Detector)	N/A	1/Valve
17.	Safety Valve Position Indicator (Backup Detector)	N/A	N/A
18.	Emergency Feedwater Flow	0-850 gpm	2/steam generator

TABLE 4.3-7
POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Power Range Nuclear Flux	M	Q*
2.	Reactor Building Pressure	M	R
3.	Source Range Nuclear Flux	M	R*
4.	Reactor Coolant Outlet Temperature	M	R
5.	Reactor Coolant Total Flow Rate	M	R
6.	RC Loop Pressure	M	R
7.	Pressurizer Level	M	R
8.	Steam Generator Outlet Pressure	M	R
9.	Steam Generator Level	M	R
10.	Borated Water Storage Tank Level	M	R
11.	Startup Feedwater Flow Rate	M	R
12.	Reactor Coolant System Subcooling Margin Monitor	M	R
13.	PORV Position Indicator (Primary Detector)	M	R
14.	PORV Position Indicator (Backup Detector)	M	R
15.	PORV Block Valve Position Indicator	M	R
16.	Safety Valve Position Indicator (Primary Detector)	M	R
17.	Safety Valve Position Indicator (Backup Detector)	M	R
18.	Emergency Feedwater Flow	M	R

*Neutron detectors may be excluded from CHANNEL CALIBRATION

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each valve (manual, power operated, or automatic) in the flow path 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 and the motor driven pump are capable of starting automatically:

a. Upon receipt of an emergency feedwater actuation OTSG A or B level low-low test signal, and

b. Upon receipt of an emergency feedwater actuation main feedwater pump turbines A and B control oil low test signal, and

c. Upon receipt of an emergency feedwater actuation OTSG A or B low pressure test signal, and

d. Upon receipt of an emergency feedwater actuation all reactor coolant pumps tripped test signal.

*c. Prior to startup after any refueling outage or other cold shutdown of longer than 30 days, verify the operability of an emergency feedwater flow path by utilizing an Emergency Feedwater Pump to pump water from the emergency feedwater supply tank to the steam generators.

*This surveillance requirement becomes effective after completion of the new emergency feedwater supply tank.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

INTRODUCTION

On September 27, 1982, the NRC staff issued a Safety Evaluation (SE) on Auxiliary (emergency) Feedwater System Automatic Initiation and Flow Indication (TMI Action Plan Item II.E.1.2). On May 1, 1984, the NRC staff issued an SE on the EFW reliability upgrade (II.E.1.1). We requested the licensee to propose necessary changes to the Technical Specifications at a time that is appropriate to the installation and operation of the new Emergency Feedwater System (EFWS). By letter dated January 23, 1985, the licensee requested a revision to the Technical Specifications to incorporate the required EFWS upgrades including operability and surveillance requirements for the emergency feedwater initiation and control logic. By letter dated June 6, 1985, the licensee submitted a letter to correct some administrative errors in the January submittal, and by letter of June 28, 1985 withdrew one change requested in the earlier submittal and clarified others.

EVALUATION AND DISCUSSION

The licensee proposed a revision to Technical Specification 3.3.2.1 and 3.3.3.6 to provide operability and surveillance requirements for the upgraded emergency feedwater system. We have reviewed the licensee's submittal and conclude that the Technical Specifications appropriately address the operability and surveillance requirements for EFWS manual initiation and automatic actuation logic and are in conformance with the B&W Standard Technical Specifications. The proposed changes are therefore acceptable.

The licensee also proposed changes to the EFWS Technical Specification 3.7.1.2 to incorporate the required EFWS upgrades. Specifically, this Technical Specification was revised to 1) delete the surveillance requirements on the air accumulators for valves FWV-39 and 40, 2) add 18-month surveillance on the newly incorporated EFW automatic actuation signals and 3) add EFW flow path operability verification following a refueling outage or other cold shutdown of longer than 30 days.

Surveillance on the air accumulators for valves FWV-39 and 40 has been deleted due to EFWS piping modifications which change the function of these valves. Prior to the modification, these valves provided EFW flow control when the EFW block valves are closed. This function is now provided by four new valves, EFV-55, 56, 57 and 58. Therefore, verification of the operability of valves FWV-39 and 40 is no longer necessary as they are no longer in the system flowpath.

Surveillance at least once per 18 months during shutdown to confirm automatic actuation of the EFWS pumps on receipt of a once through steam generator (OTSG) A or B low pressure test signal and all reactor coolant pumps tripped test signal has been incorporated. These automatic initiation signals were added in order to improve EFWS availability to mitigate certain transient conditions. This surveillance frequency is the same as that for the previously existing automatic actuation signals.

Per the recommendations from the NUREG-0737 Item II.E.1.1 review, the licensee has added a Technical Specification change to assure the operability of an EFWS flowpath by using an EFW pump to pump water from the water supply tank to the steam generators prior to startup after any refueling outage or other cold shutdown of 30 days or longer. This change satisfies the concern that the EFWS be in its operational configuration during power operation when it has not been utilized for plant startup. However, because the current EFW water source (condensate storage tank) is not condensate quality and may therefore cause degradation of steam generator internals, the licensee proposed not to implement this change until after installation of the new emergency feedwater supply tank during the Cycle 6 refueling outage. We find this schedule to be acceptable.

Based on our review of the proposed changes to Technical Specification 3.7.1.2 and a comparison to the Safety Evaluation for Item II.E.1.1 of NUREG-0737, we conclude that the EFWS Technical Specification changes are in accordance with our criteria for improving EFWS reliability and are therefore acceptable.

In its letter of June 28, 1985, the licensee withdrew a previously requested change to the feedwater isolation response time (Table 3.3-5) leaving it unchanged from the present Technical Specifications because the requested change was overly conservative. The licensee committed to reevaluate this response time, and if a change is indicated, to submit an amendment request to reflect the appropriate response time. We find this acceptable.

Various word changes of an administrative nature are proposed to clarify existing requirements. Because these do not change the original meaning and intent of the Technical Specifications, these are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 16, 1985

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