Mr. John Paul Cowan
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing (NA1B)
Crystal River Energy Complex
15760 W. Power Line Street
Crystal River, Florida 34428-6708

May 21, 1999

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING REACTOR PROTECTION SYSTEM AND ENGINEERED SAFEGUARDS ACTUATION SYSTEM SETPOINTS AND SURVEILLANCE REQUIREMENTS (TAC NO. MA3614)

Dear Mr. Cowan:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 178 to Facility Operating License No. DPR-72 for Crystal River Unit 3. This amendment is in response to a Florida Power Company (FPC) request dated November 23, 1998, and supplemented on January 29 and May 7, 1999. FPC proposed to change the Improved Technical Specifications (ITSs) for the reactor protection system and engineered safeguards actuation system. A change to a surveillance requirement to verify valve position in the high pressure injection (HPI) system was also proposed. These ITS changes are intended, in part, to reduce the operator actions necessary to mitigate certain small-break loss-of-coolant accidents. Included as background in the FPC submittal was a description of planned modifications to the high pressure injection (HPI) system. FPC confirmed in the January 29, 1999 letter that these modifications will be evaluated in accordance with 10 CFR 50.59, and NRC review and approval for these modifications was not being requested.

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

Sincerely,

Original signed by:

L. Wiens, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures: 1. Amendment No. 178 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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

May 21, 1999

Mr. John Paul Cowan
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing (NA1B)
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L. Wiens, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
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Docket No. 50-302

Enclosures: 1. Amendment No. 178 to DPR-72
2. Safety Evaluation
cc w/enclosures: See next page

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
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. 178
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 November 23, 1998, as supplemented on January 29 and May 7, 1999, 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. 178 , 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 and shall be implemented prior to commencing Cycle 12 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Sheri R. Peterson, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Date of Issuance: May 21, 1999

ATTACHMENT TO LICENSE AMENDMENT NO.

TO FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contains marginal lines indicating the areas of change. *Indicates spillover page.

Remove Page	Insert Page
3.3-5	3.3-5
3.3-13	3.3-13
3.3-15	3.3-15
3.5-5	3.5-5
B 3.3-8	B 3.3-8
B 3.3-9	B 3.3-9
B 3.3-20	B 3.3-20
B 3.3-21	B 3.3-21
B 3.3-46	B 3.3-46
B 3.3-47	B 3.3-47
B 3.3-51	B 3.3-51
B 3.3-52	B 3.3-52
B 3.3-54	B 3.3-54
B 3.3-125A	B 3.3-125A
B 3.3-129	B 3.3-129
B 3.3-130	B 3.3-130
B 3.5-12	B 3.5-12
B 3.5-13	B 3.5-13
B 3.5-18	B 3.5-18
B 3.5-19	B 3.5-19*

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower –				
a. High Setpoint	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 SR 3.3.1.7	≤ 104.9% RTP
b. Low Setpoint	2[b],3[b] 4[b],5[b]	G	SR 3.3.1.1 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 618°F
3. RCS High Pressure	1,2	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	≤ 2355 psig
4. RCS Low Pressure	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	≥ 1900 psig
5. RCS Variable Low Pressure	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ (11.59 * T_{hot} - 5037.8) psig
6. Reactor Building High Pressure	1,2,3[c]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 4 psig
7. Reactor Coolant Pump Power Monitor (RCPPM)	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	More than one pump drawing ≤ 1152 or ≥ 14,400 kW
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2[a]	F	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE setpoint envelope in COLR
9. Main Turbine Trip (Control Oil Pressure)	≥ 45% RTP	H	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 45 psig
10. Loss of Both Main Feedwater Pumps (Control Oil Pressure)	≥ 20% RTP	I	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 55 psig
11. Shutdown Bypass RCS High Pressure	2[b],3[b] 4[b],5[b]	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 1820 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD Control System (CRDCS) capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRDCS capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2	--------NOTE-------- Only required for RCS Pressure—Low Parameter. ---------- Reduce RCS pressure < 1800 psig.	 12 hours
	AND		
	C.3	--------NOTE-------- Only required for RCS Pressure—Low Low Parameter. ---------- Reduce RCS pressure < 900 psig.	 12 hours
	AND		
	C.4	--------NOTE-------- Only required for Reactor Building Pressure High setpoint and High High Parameter. ---------- Be in MODE 4.	 12 hours

Table 3.3.5-1 (page 1 of 1)
Engineered Safeguards Actuation System Instrumentation

PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Reactor Coolant System Pressure – Low	≥ 1800 psig	≥ 1625 psig
2. Reactor Coolant System Pressure – Low Low	≥ 900 psig	≥ 500 psig
3. Reactor Building Pressure – High	1,2,3	≤ 4 psig
4. Reactor Building Pressure – High High	1,2,3	≤ 30 psig

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.2	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.3	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.4	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.5.2.5	Verify the following valves in the HPI flow path are locked, sealed or otherwise secured in the correct position: a. MUV-2; b. MUV-6; c. MUV-10; d. MUV-590; e. MUV-591; f. MUV-592; and g. MUV-593.	24 months

(continued)

BASES

BACKGROUND

Channel Bypass (continued)

contacts from the other channels with the channel bypass relay. If any contact is open, the second channel cannot be bypassed. The second condition is the closing of the key switch. When the bypass relay is energized, the bypass contact closes, maintaining the channel trip relay in an energized condition. All RPS trip logics are reduced to a two-out-of-three logic in channel bypass.

Shutdown Bypass

During plant cooldown, it is desirable to maintain the safety rods withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.). However, if the safety rods are withdrawn too soon following reactor shutdown as RCS pressure is decreased, an RCS Low Pressure trip will occur at 1900 psig and the rods will re-insert into the core. To avoid this, the protection system allows the operator to bypass the low pressure trip and maintain shutdown capabilities.

During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip of 1900 psig. The RCS pressure is decreased to less than 1820 psig, then each RPS channel is placed in shutdown bypass.

In shutdown bypass, a normally closed contact opens and the operator closes the shutdown bypass key switch in each RPS channel. This action bypasses the RCS Low Pressure trip, Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Reactor Coolant Pump overpower/underpower trip, and the RCS Variable Low Pressure trip, and inserts a new RCS High Pressure, 1820 psig trip. The operator can now withdraw the safety rods for additional available reactivity insertion.

The insertion of the new high pressure trip performs two functions. First, with a trip setpoint of 1820 psig, the bistable prevents operation at normal system pressure, 2155 psig, with a portion of the RPS bypassed. The second function is to ensure that the bypass is removed prior to normal operation. When the RCS pressure is increased during

(continued)

BASES

BACKGROUND

Shutdown Bypass (continued)

a plant heatup, the safety rods are inserted prior to reaching 1820 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1900 psig. The safety rods can then be withdrawn and remain at the full out condition for the rest of the heatup.

In addition to the Shutdown Bypass RCS High Pressure trip, the nuclear overpower high flux trip setpoint is administratively reduced to 5% RTP while the RPS is in shutdown bypass. This provides a backup to the Shutdown Bypass RCS High Pressure trip and allows low temperature physics testing while preventing the generation of any significant amount of power.

Module Interlock and Test/Interlock Trip Relay

Each channel and each trip module is capable of being individually tested. When a module is placed into the test mode or is removed from the system, it causes the test/interlock trip relay to de-energize and to indicate an RPS channel trip. Under normal conditions, the channel to be tested is placed in bypass before a module is tested. This ensures the channel trip relay remains energized during testing and the channel does not trip.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Chapter 14 of the FSAR takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the safety evaluation report (SER) written for the CR-3 operating license. Functions not specifically credited include high RB pressure, high RCS temperature, main turbine trip, shutdown bypass-RCS pressure high, and loss of both main feedwater pumps.

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ 1820 psig and the Nuclear Overpower—Low Setpoint set at or below 5% RTP, the trips listed below can be bypassed. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower—Low Setpoint trip prevent conditions from reaching a point where actuation of these Functions would be required.

1.a Nuclear Overpower—High Setpoint;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump Power Monitors; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

The Shutdown Bypass RCS High Pressure Function's Allowable Value is selected to ensure a trip occurs before producing THERMAL POWER.

The RPS satisfies Criterion 3 of the NRC Policy Statement.

In MODES 1 and 2, the following trips shall be OPERABLE. These trips are designed to rapidly make the reactor subcritical in order to protect the SLs during AOOs and to function along with the ESAS to provide acceptable consequences during accidents.

1.a Nuclear Overpower—High Setpoint;

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

7. Reactor Coolant Pump Over/Under Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

Functions 1, 4, 5, 7, and 8 may be bypassed in MODE 2 or below (higher numerical MODE) when RCS pressure is below 1820 psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower—Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower—Low setpoint trip prevent conditions from reaching a point where actuation of these Functions is necessary.

Two other Functions are required to be OPERABLE during portions of MODE 1. These are the Main Turbine Trip (Control Oil Pressure) and the Loss of Main Feedwater Pumps (Control Oil Pressure) trip. These Functions are required to be OPERABLE above 45% RTP and 20% RTP, respectively. Analyses presented in BAW-1893 (Ref. 5) showed that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 4).

Because the only safety function of the RPS is to interrupt power to the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5 if the reactor trip breakers are open, or the CRDCS is incapable of rod withdrawal.

Similarly, the RPS is not required to be OPERABLE in MODE 6 when the CONTROL RODS are decoupled from the CRDs. However, in MODE 2, 3, 4, or 5, the Shutdown Bypass RCS High Pressure and Nuclear Overpower—Low Setpoint trip Functions are required to be OPERABLE if the CRD trip breakers are closed and the CRDCS is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower—Low setpoint trips are sufficient to prevent an approach to conditions that could challenge SLs.

(continued)

BASES

BACKGROUND (continued)

related safeguards equipment will not inhibit the overall ES Functions. Where a motor operated or a solenoid operated valve is driven by either of two matrices, one is from actuation train A and one from actuation train B. Redundant ES pumps are controlled from separate and independent actuation channels.

Engineered Safety Feature Actuation System Bypasses

No provisions are made for maintenance bypass of ESAS instrumentation channels. Operational bypasses are provided, as discussed below, to allow accident recovery actions to continue and, to allow plant cooldown without spurious ESAS actuation.

The ESAS RCS pressure instrumentation channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low and low low pressure trips are required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are no longer applicable.

No more than two (of the three) High RB Pressure channels may be manually bypassed after an actuation. The manual bypass allows operators to take manual control of ES Functions after initiation to allow recovery actions.

Reactor Coolant System Pressure

RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that provide an input to the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by four bistables to provide two trip signals, at 1625 psig and 500 psig, and two bypass permissive signals, at 1800 psig and 900 psig.

(continued)

BASES

BACKGROUND

Reactor Coolant System Pressure (continued)

The outputs of the three channels trip bistables, associated with the low RCS pressure (1625 psig) actuate bistable trip auxiliary relays in two sets (actuation trains A and B) of identical and independent trains. The two HPI trains each use three logic channels arranged in two-out-of-three coincidence networks. The outputs of the three bistables associated with the Low Low RCS Pressure (500 psig) actuate bistable trip auxiliary relays in two sets (actuation trains A and B) of identical and independent trains. The two LPI trains each use three logic channels arranged in two-out-of-three coincidence networks for LPI Actuation. The outputs of the three Low Low RCS Pressure bistables also trip the automatic actuation relays, via a LPI bistable trip auxiliary relay, in the corresponding HPI train as previously described.

Reactor Building Pressure

ESAS RB pressure signal information is provided by 12 pressure switches. Six pressure switches are used for the High RB Pressure Parameter, and six pressure switches are used for the High-High RB Pressure Parameter.

The output contacts of six High RB Pressure switches are used in two sets of identical and independent actuation trains. These two trains each use three logic channels. The outputs of these channels are used in two-out-of-three coincidence networks. The output contacts of the six RB pressure switches also trip, via a pressure switch trip auxiliary relay, the automatic actuation relays in the corresponding HPI and LPI trains as previously described.

The output contacts of six High High RB Pressure switches are used in two sets of identical and independent actuation trains. The outputs of the High High RB Pressure switches are used in two-out-of-three coincident networks for RB Spray Actuation. The two-out-of-three logic associated with each RB Spray train actuates spray pump operation when the High-High RB signal and the HPI signal are coincident in that train.

(continued)

BASES

LCO (continued)

1. Reactor Building Pressure—High Setpoint

The RB Pressure—High Setpoint Allowable Value ≤ 4 psig was selected to be low enough to detect a rise in RB Pressure that would occur due to a small break LOCA, thus ensuring that the RB high pressure actuation of the safety systems will occur for a wide spectrum of break sizes. The trip setpoint also causes the RB coolers to shift to low speed (performed as part of the HPI logic) to prevent damage to the cooler fans due to the increase in the density of the air steam mixture present in the containment following a LOCA.

2. Reactor Building Pressure—High High Setpoint

The RB Pressure—High High Setpoint Allowable Value ≤ 30 psig was chosen to be high enough to avoid actuation during an SLB, but also low enough to ensure a timely actuation during a large break LOCA.

APPLICABILITY

The ESAS instrumentation for each Parameter is required to be OPERABLE during the following MODES and specified conditions.

1. Reactor Coolant System Pressure—Low Setpoint

The RCS Pressure—Low Setpoint actuation Parameter shall be OPERABLE during operation above 1800 psig. This ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or SLB. Below 1800 psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal cooldown when safety system actuations are not required.

The allowance for the bypass is consistent with the plant transition to a lower energy state, providing greater margins to core and containment limits. The response to any event, given that the reactor is already shut down, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during plant heatup from an outage when the RCS energy content is low.

(continued)

BASES

APPLICABILITY

1. Reactor Coolant System Pressure—Low Setpoint (continued)

To ensure the RCS Pressure—Low trip is not bypassed when required to be OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with a setpoint of ≤ 1800 psig. The bypass removal does not need to function for accidents initiated from RCS Pressures below the bypass removal setpoint.

2. Reactor Coolant System Pressure—Low Low Setpoint

The RCS Pressure—Low Low Setpoint actuation Parameter shall be OPERABLE during operation above 900 psig. This ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA. Below 900 psig, the low low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal plant cooldown.

The allowance for the bypass is consistent with plant transition to a lower energy state, providing greater margins to core and containment limits. The response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during heatup from an outage when the RCS energy content is low.

To ensure the RCS Pressure—Low Low trip is not bypassed when assumed OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with a setpoint of ≤ 900 psig. The bypass removal does not need to function for accidents initiated by RCS Pressure below the bypass removal setpoint.

(continued)

BASES

ACTIONS (continued)

B.1

Condition B applies when one required instrumentation channel in one or more RB Pressure Parameters becomes inoperable. If one required channel is inoperable, placing it in a tripped Condition leaves the affected actuation train in one-out-of-one condition for actuation and the other actuation channel in a two-out-of-two condition (making the worst case assumption the third channel in each actuation train is not OPERABLE). In this condition, if another RB Pressure ESAS channel were to fail, the ESAS instrumentation could still perform its actuation function. For RB Pressure Parameters, all affected pressure switch trip auxiliary relays must be tripped to comply with this Required Action. This is normally accomplished by tripping the affected pressure switch test switch.

The 72 hour Completion Time is based on engineering judgment and is sufficient time to perform the Required Action.

C.1, C.2, C.3, and C.4

If Required Actions A.1 or B.1 cannot be met within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and, for the RCS Pressure—Low Parameter, to < 1800 psig, for the RCS Pressure—Low Low Parameter, to < 900 psig, and for the RB Pressure High Parameter and High High Parameter, to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

All ESAS Parameter instrumentation listed in Table 3.3.5-1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing.

(continued)

BASES

The following table identifies the specific instrument tag numbers for PAM instrumentation identified in Table 3.3.17-1.

FUNCTION	CHANNEL A	CHANNEL B
1. Wide Range Neutron Flux	NI-15-NI-1 or NI-15-NIR	NI-14-NI-1
2. RCS Hot Leg Temperature	RC-4A-TI4-1	RC-4B-TIR1
3. RCS Pressure(Wide Range)	RC-158-PI2 or RC-158-PIR	RC-159-PI2
4. Reactor Coolant Inventory	RC-163A-LR1 (Hot leg level) and RC-164A-LR1 (Vessel Head level)	RC-163B-LR1 (Hot leg level) and RC-164B-LR1 (Vessel Head level)
5. Borated Water Storage Tank Level	DH-7-LI or DH-7-LIR1	DH-37-LI
6. High Pressure Injection Flow	A1: MU-23-FI8-1 A2: MU-23-FI10 B1: MU-23-FI9 B2: MU-23-FI7-1	A1: MU-23-FI12 A2: MU-23-FI6-1 B1: MU-23-FI5-1 B2: MU-23-FI11
7. Containment Sump Water Level (Flood Level)	WD-303-LI or WD-303-LR	WD-304-LI or WD-304-LR
8. Containment Pressure (Expected Post-Accident Range)	BS-16-PI	BS-17-PI
9. Containment Pressure (Wide Range)	BS-90-PI or BS-90-PR	BS-91-PI or BS-91-PR
10. Containment Isolation Valve Position	**ES Light Matrix "A":** AHV-1B/1C; CAV-1/3/4/5/126/429/430/433/434; CFV-11/12/15/16; LRV-70/72; MUV-258 thru -261/567;WDV-3/60/94/406; WSV-3/5/28 thru -31/34/35/42/43	**ES Light Matrix "B":** AHV-1A/1D; CAV-2/6/7/431/432/435/436; CFV-29/42;LRV-71/73;MUV-18/27/49/253; WDV-4/61/62/405; WSV-4/6/26/27/32/33/38/39/40/41
	ES Light Matrix "AB": CFV-25 thru-28; CIV-34/35/40/41; DWV-160; MSV-130/148; SWV-47 thru 50/79 thru 86/109/110	
11. Containment Area Radiation (High Range)	RM-G29-RI or RM-G29-RIR	RM-G30-RI
12. Containment Hydrogen Concentration	WS-11-CR	WS-10-CR
13. Pressurizer Level	RC-1-LIR-1	RC-1-LIR-3
14. Steam Generator Water Level (Startup Range)	**OTSG A:** SP-25-LI1 or SP-25-LIR **OTSG B:** SP-29-LI1 or SP-29-LIR	**OTSG A:** SP-26-LI1 **OTSG B:** SP-30-LI1

(continued)

BASES

LCO (continued)

5. Borated Water Storage Tank (BWST) Level

BWST inventory is monitored by level instrumentation with a span of 0 to 50 feet. Redundant monitoring capability is provided by three independent level measurements. Two level transmitters provide input to control room indicators, and one of these channels is recorded in the control room. The control room indications are the primary indications used by the operator. Therefore, the LCO deals specifically with this portion of the instrument string.

During a design basis LOCA, the Reactor Building Spray, Low Pressure Injection (LPI) and High Pressure Injection (HPI) Systems are automatically aligned to obtain suction from the BWST. As the BWST inventory is pumped into the RCS and containment, coolant will be lost through the break and will accumulate in the reactor building sump. The operator is required to switch LPI and RB Spray suction to the reactor building emergency sump from the BWST when the BWST level reaches a specified level setpoint. At this same time if the RCS pressure is greater than the LPI pump shutoff head, it will also be necessary to switch the suction of the HPI pumps to the discharge of the LPI pumps to ensure the capability to inject flow to the RCS since the HPI pumps do not have the capability of drawing coolant from the sump. BWST level is a Type A variable because it is the primary indication used by the operator to determine when to initiate the switch-over to sump recirculation. This operator action is necessary to satisfy the long-term core cooling requirements specified in 10 CFR 50.46.

6. HPI Flow (Low Range)

HPI flow instrumentation is provided for verification and long term monitoring of HPI flow. HPI flow is determined from differential pressure transmitters. Two channels in each of the four injection lines, for a total of eight low range indicators, provides this indication. One transmitter is calibrated to a range of 0-200 gpm. Each differential pressure measurement provides an input to a control room indicator. Since the operator relies on the control room indication following an accident, the LCO deals with this portion of the instrument string.

(continued)

BASES

LCO

6. HPI Flow (Low Range) (continued)

Although 4 high range flow indicators (0-500 gpm) readout on the main control board, they are not used to accomplish any safety functions. The HPI lines will have preset throttle valves, stop check valves, and crosstie lines to: (1) create the desired flow distribution through the HPI lines for LOCA core cooling; (2) ensure adequate cooling flow to the HPI pump mechanical seals; and (3) prevent HPI pump flow from exceeding 600 gpm (maximum HPI pump flow rate assumed in design calculations associated with Emergency Diesel Generator loading, ECCS pump available NPSH, and makeup tank (MUT-1) allowable overpressure versus level).

7. Containment Sump Water Level (Flood Level)

Containment sump water level (Flood) is monitored by two channels of level indication, both of which are displayed in the control room on edgewise level indicators. Channel A and B sump flood level indication are recorded in the associated 'A' and 'B' EFIC Rooms. Each instrument encompasses a range of 0-10 feet above the sump and provides information to the operator related to gross leakage in the Reactor Building. This leakage may be indication of degradation in the reactor coolant pressure boundary (RCPB) which would require further investigation and action. These instruments are not assumed to provide information required by the operator to take a mitigation action specified in the accident analysis. As such, they are not Type A variables. However, the monitors are deemed risk significant (Category 1) and are included within the LCO based upon this consideration.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or CONTROL ROD assembly insertion for small breaks. Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core.

The LCO ensures that an ECCS train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core uncovery for a large break LOCA. It also ensures that the HPI pump will deliver sufficient water for a small break LOCA and provide sufficient boron to maintain the core subcritical following the small break LOCA or an SLB.

In the LOCA analyses, HPI and LPI are not credited until 35 seconds after actuation of the ESAS signal. This is based on a loss of offsite power and the associated time delays in startup and loading of the emergency diesel generator (EDG). Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all.

The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that at least one is available, assuming a single active failure in the other train. With one ECCS train inoperable, the system is still capable of mitigating an event, providing a concurrent single failure does not occur. Hence, the 72 hour ACTION addressing a loss of redundancy is appropriate.

(continued)

BASES

LCO (continued)

Not all portions of the HPI flow path satisfy the independence criteria discussed above. Specifically, the HPI flow path downstream of the HPI/Makeup pumps is not separable into two distinct trains, and is therefore, not independent. This conclusion is based upon analysis which shows, that in the event of a postulated break in the HPI injection piping, injection flow is required through a minimum of three (3) injection legs, assuming one pump operation, or through a minimum of two (2) injection legs, assuming two HPI pump operation. When considering the impact of inoperabilities in this portion of the system, the same concept of maintaining single active failure protection must be applied. When components become inoperable, an assessment of the HPI systems ability to perform its safety function must be performed. If the system can continue to perform its safety function, without assuming a single active failure, then the 72 hour loss of redundancy ACTION is appropriate. If the inoperability renders the system, as is, incapable of performing its safety function, without postulating a single active failure, then the plant is in a condition outside the safety analysis and must enter LCO 3.0.3 immediately.

In MODES 1, 2, and 3, an ECCS train consists of an HPI subsystem and an LPI subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ESAS signal and manually transferring suction to the reactor building emergency sump.

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI and LPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow path may be manually transferred to take its supply from the reactor building emergency sump and to supply its flow to the RCS via two paths, as described in the Background section.

The flow path for each train must maintain its designed degree of independence to ensure that no single active failure can disable both ECCS trains.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5

Verification of the positions of the listed valves in the HPI flowpath ensures adequate flow resistance in the overall system and the individual HPI lines. Maintenance of adequate flow resistance and pressure drop in the piping system for each injection point is necessary in order to: (1) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS LOCA analyses; (2) provide an acceptable level of total ECCS flow to all injection points equal to or above values assumed in the ECCS LOCA analyses; (3) ensure adequate cooling flow to the HPI pump mechanical seals; and (4) prevent HPI pump flow from exceeding 600 gpm when the system is in its minimum resistance configuration (600 gpm is the maximum HPI pump flow rate assumed in design calculations associated with Emergency Diesel Generator loading, ECCS pump available NPSH, and makeup tank (MUT-1) allowable overpressure versus level). This 24 month Frequency is acceptable based on consideration of the design reliability of valves that are locked, sealed, or otherwise secured in position.

Verification of correct valve position will be accomplished by assuring the mechanism that locks, seals or secures the valves is intact. If the stop check valves or throttle valves are repositioned, the valves must be returned to their correct position and then secured. This "as-left" position verification ensures the HPI flow assumptions in the accident analysis are maintained.

SR 3.5.2.6

This Surveillance ensures that the flow controllers for the LPI throttle valves will automatically control the LPI train flow rate in the desired range and prevent LPI pump runout as RCS pressure decreases after a LOCA. The 24 month Frequency is acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

Periodic inspections of the reactor building emergency sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and to preserve access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

REFERENCES

1. 10 CFR 50.46.

2. FSAR, Section 6.1.

3. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

4. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWP-3000.

5. FTI 51-1266138-01, Safety Analysis Input to Startup Team Safety Assessment.

6. FSAR, Section 4.3.10.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. DPR-72

REACTOR PROTECTION SYSTEM AND ENGINEERED SAFEGUARDS SYSTEM

ACTUATION SYSTEM SETPOINTS AND SURVEILLANCE REQUIREMENTS

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

1.0 INTRODUCTION

By letters dated November 23, 1998, and supplemented January 29 and May 7, 1999, Florida Power Corporation (FPC or the licensee) requested the U. S. Nuclear Regulatory Commission (NRC) approval for changes to the technical specifications (TSs) for the reactor protection system (RPS) and engineered safeguards actuation system (ESAS) setpoints and allowable values. The licensee has also proposed a change to a surveillance requirement to verify valve position for valves in the high pressure injection (HPI) system. The licensee's Small Break Loss of Coolant Accident (SBLOCA) analysis for Crystal River Unit 3 (CR-3) currently takes credit for several operator actions in order to achieve a successful mitigation. In 1997, FPC committed to enhance the HPI system before Cycle 12 in order to reduce the number of operator actions required to mitigate SBLOCA. The proposed modifications are intended to provide a significant reduction in operator burden and enhance the operator's capabilities for accident management for design basis accidents, specifically a SBLOCA concurrent with a loss of offsite power and a single failure, such as a loss of "A" train DC electrical system or battery.

The licensee has also included, as supporting information, a description of a number of changes to the HPI system which are intended to be implemented under authority of 10 CFR 50.59. The staff confirmed with the licensee that modifications to the HPI system were not part of the license amendment request and NRC review and approval for these modifications were not requested. Therefore, the modifications to the HPI system were not reviewed for acceptability.

The January 29, 1999 and May 7, 1999, supplements did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the Federal Register.

2.0 EVALUATION

2.1 Setpoint Calculation Methodology

Revised Allowable Values were determined using the licensee's in-house setpoint calculation methodology. This methodology was reviewed and accepted by the staff via NRC Inspection of CR-3 and was documented in Inspection Report 50-302 / 95-06 dated March 1, 1995. Use of this methodology to determine the RPS and ESAS setpoint allowable values is acceptable.

2.2 TS Table 3.3.1-1, Items 4 and 11 Reactor Protection System Instrumentation

The licensee is raising the allowable value for the RPS reactor trip setpoint for Reactor Coolant System (RCS) low pressure from 1800 to 1900 psig. The allowable value for the RPS setpoint for Shutdown Bypass RCS High Pressure is also being raised from 1720 to 1820 psig. These changes are being made to maintain the existing relationship between the reactor trip, the reactor trip bypass, the ESAS signal and ESAS applicability on low RCS pressure. Because the ESAS setpoint is being raised, it is desirable to also raise the reactor low pressure and shutdown bypass high pressure trips.

Although increasing the low-pressure reactor trip setpoint may increase the frequency of reactor trips, the licensee has reviewed data on reactor trips at Babcock and Wilcox (B&W) plants since 1980 and concluded that the increase is not significant (approximately one percent). The staff finds this conclusion reasonable and finds the proposed change acceptable. The revised setpoints will ensure earlier actuation of the RPS and ESAS on a low RCS pressure condition and no reduction in the margin of safety associated with these systems is effected by the proposed change. The Shutdown Bypass RCS Pressure High trip is a backup for mitigating shutdown transients and is not credited in the transient analysis. As a result, raising this allowable value is also acceptable.

2.3 TS 3.3.5 and Table 3.3.5-1 Engineered Safeguards Actuation System Instrumentation

The licensee is raising the allowable value for the ESAS RCS Pressure - Low from 1500 to 1625 psig. The signal actuates the HPI system on decreasing RCS pressure and the proposed change, raising the setpoint, will result in earlier actuation of HPI on decreasing RCS pressure. For a small-break LOCA, where the event progresses slowly this change will result in HPI being actuated significantly earlier with more water delivered to the core. The change enhances the plant response for some slow-progressing events by increasing the integrated flow of water to the core. As a result, the staff finds the proposed change acceptable.

Additionally, to permit this change the applicability for ESAS in TS table 3.3.5-1 is also being adjusted from 1700 to 1800 psig. This permits the ESAS to be bypassed below 1800 psig when depressurizing well before the setpoint is reached. Bypassing the ESAS well before the setpoint reduces the likelihood of an inadvertent ESAS actuation while shutting down and depressurizing. The same change is also being made to Action C.2 in TS 3.3.5 which requires RCS pressure to be reduced to 1800 psig rather than 1700 psig. Because inadvertent ESAS actuations can be detrimental, the staff finds the change acceptable.

2.4 TS 3.5.2.5 Surveillance Requirement

TS Surveillance Requirement (SR) 3.5.2.5 is the SR used to verify valve position for valves in the HPI flow path, and is being revised to include the additional throttle valves to be installed in the system that need to be verified in the correct position on a 24-month interval. The installation of these valves is part of the planned system modification to the system. Because there are currently valves in the system that have an SR to verify their position, the licensee is proposing to include these throttle valves under the current surveillance. The proposed SR 3.5.2.5 has been modified to specifically call out the seven valves that would be verified in the correct position, MUVs 2, 6, 10, 590, 591, 592 and 593. The revision to the surveillance is consistent with the intent of the B&W Owners Group Standard Technical Specifications and provides additional assurance that the valves will be in the correct position when the system is needed. As a result, the staff finds the proposed change acceptable.

3.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. NRC, the State of Florida does not desire notification of issuance of license amendments.

4.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (63 FR 71966). Accordingly, the 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 the amendment.

5.0 CONCLUSION

The staff has reviewed the TS amendment request, the technical justification, and the applicable TS bases submitted by the licensee. The staff finds the requested changes to the RPS and ESAS Allowable Values and the change to SR 3.5.2.5 are acceptable. The staff concludes that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: May 21, 1999

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CRYSTAL RIVER UNIT NO. 3

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