December 22, 1997

Mr. Roy A. Anderson
Senior Vice President,
Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
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15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - STAFF EVALUATION AND ISSUANCE OF AMENDMENT RE: POST-ACCIDENT MONITORING INSTRUMENTATION (TAC NO. M99308)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 162 to Facility Operating License No. DPR-72 for the Crystal River Unit 3. The amendment consists of changes to the existing Technical Specifications (TS) in response to your request dated July 29, 1997, as supplemented by letter dated October 29, 1997. Florida Power Corporation requested an amendment to the post-accident monitoring instrumentation TS.

The clarifying information provided by your October 29, 1997, letter did not affect the U.S. Nuclear Regulatory Commission's original proposed no significant hazards consideration.

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. Raghavan, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-302

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Enclosures:
1. Amendment No. 162 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 22, 1997

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Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 162 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 162
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 29,1997, as supplemented on October 29, 1997, 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.

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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. 162, 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 22, 1997

ATTACHMENT TO LICENSE AMENDMENT NO.162

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 contain vertical lines indicating the area of change.

Remove	Replace
3.3-41	3.3-41
B3.3-126	B3.3-126
B3.3-127	B3.3-127
B3.3-131	B3.3-131
B3.3-138	B3.3-138
-	B3.3-138A
-	B3.3-138B

Table 3.3.17-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Wide Range Neutron Flux	2	E
2. RCS Hot Leg Temperature	2	E
3. RCS Pressure (Wide Range)	2	E
4. Reactor Coolant Inventory	2	F
5. Borated Water Storage Tank Level	2	E
6. High Pressure Injection Flow	2 per injection line	E
7. Containment Sump Water Level (Flood Level)	2	E
8. Containment Pressure (Expected Post-Accident Range)	2	E
9. Containment Pressure (Wide Range)	2	E
10. Containment Isolation Valve Position	2 per penetration[(a)(b)]	E
11. Containment Area Radiation (High Range)	2	F
12. Containment Hydrogen Concentration	2	E
13. Pressurizer Level	2	E
14. Steam Generator Water Level (Start-up Range)	2 per OTSG	E
15. Steam Generator Water Level (Operating Range)	2 per OTSG	E
16. Steam Generator Pressure	2 per OTSG	E
17. Emergency Feedwater Tank Level	2	E
18. Core Exit Temperature (Backup)	3 per core quadrant	E
19. Emergency Feedwater Flow	2 per OTSG	E
20. Low Pressure Injection Flow	2	E
21. Degrees of Subcooling	2(d)	E
22. Emergency Diesel Generator kW Indication	2[(c)]	E

(a) Only one position indication is required for penetrations with one Control Room indicator.

(b) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(c) One indicator per EDG.

(d) These two channels of subcooling margin are backed up by either of two indications of subcooling margin based on similar inputs through the Safety Parameter Display System (SPDS). At least one SPDS channel must be available to provide this backup. With both SPDS channels INOPERABLE, Condition C is applicable.

BASES

LCO (continued)

The following list is a discussion of the specified instrument Functions listed in Table 3.3.17-1.

1. Wide Range Neutron Flux

Two wide-range neutron flux monitors are provided for post-accident reactivity monitoring over the entire range of expected conditions. Each monitor provides indication over the range of 10^{-8} to 100% log rated power covering the source, intermediate, and power ranges. Each monitor utilizes a fission chamber neutron detector to provide redundant main control board indication. A single channel provides recorded information in the control room. The control room indication of neutron flux is considered one of the primary indications used by the operator following an accident. Following an event the neutron flux is monitored for reactivity control. The operator ensures that the reactor trips as necessary and that emergency boration is initiated if required. Since the operator relies upon this indication in order to take specified manual action, the variable is included in this LCO. Therefore, the LCO deals specifically with this portion of the string.

2. Reactor Coolant System (RCS) Hot Leg Temperature

Two wide range resistance temperature detectors (RTD's), one per loop, provide indication of reactor coolant system hot leg temperature (T_H) over the range of 120° to 920°F. Each T_H measurement provides an input to a control room indicator. Channel B is also recorded in the control room. Since the operator relies on the control room indication following an accident, the LCO deals specifically with this portion of the string.

T_H is a Type A variable on which the operator bases manual actions required for event mitigation for which no automatic controls are provided.

(continued)

BASES

LCO

2. Reactor Coolant System (RCS) Hot Leg Temperature (continued)

Following a steam generator tube rupture, the affected steam generator is to be isolated only after T_H falls below the saturation temperature corresponding to the pressure setpoint of the main steam safety valves. For event monitoring once the RCP's are tripped, T_H is used along with the core exit temperatures and RCS cold leg temperature to measure the temperature rise across the core for verification of core cooling.

3. RCS Pressure (Wide Range)

RCS pressure is measured by pressure transmitters with a span of 0-3000 psig. Redundant monitoring capability is provided by two trains of instrumentation. Control room and remote shutdown panel indications are provided. The control room indications are the primary indications used by the operator during an accident. Therefore, the LCO deals specifically with this portion of the instrument string.

RCS pressure is a Type A variable because the operator uses this indication to adjust parameters such as steam generator (OTSG) level or pressure in order to monitor and maintain a controlled cooldown of the RCS following a steam generator tube rupture or small break LOCA. In addition, HPI flow is throttled based

(continued)

BASES

LCO (continued)

8,9. <u>Containment Pressure (Expected Post-Accident Range and Wide Range)</u>

The containment pressure variable is monitored by two ranges of pressure indication. Expected post-accident range (-10 to 70 psig) and wide range (0 to 200 psig) pressure indication each provide two channels of pressure indication. Channel A and B wide range containment pressure are recorded in the associated 'A' and 'B' EFIC Rooms. The low range is required in order to ensure instrumentation of the necessary accuracy is available to monitor conditions in the RB during DBAs. The wide range instrument was required by Regulatory Guide 1.97 to be capable of monitoring pressures over the range of atmospheric to three times containment design pressure (approximately 165 psig). Thus, it was intended to monitor the RB in the event of an accident not bounded by the plant safety analysis (i.e., a Severe Accident).

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

LCO

18. Core Exit Temperature (Backup) (continued)

following a steam generator tube rupture or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting OTSG level or pressure, would be prompted by this indication.

19. Emergency Feedwater Flow

EFW Flow instrumentation is provided to monitor operation of decay heat removal via the OTSGs. The EFW injection flow to each OTSG (2 channels per OTSG, one associated with each EFW injection line) is determined from a differential pressure measurement calibrated to a span of 0 gpm to 1000 gpm. Each differential pressure transmitter provides an input to a control room indicator and the plant computer.

EFW Flow is used by the operator to determine the need to throttle flow during accident or transient conditions to prevent the EFW pumps from operating in runout conditions or from causing excessive RCS cooldown rates when low decay heat levels are present. EFW Flow is also used by the operator to verify that the EFW System is delivering the correct flow to each OTSG. However, the primary indication of this function is provided by OTSG level.

These instruments are not assumed to provide information required by the operator to take a mitigation action specified in the safety 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)

LCO (continued)

20. Low Pressure Injection Flow

Low pressure injection flow instrumentation is provided to monitor flow to the RCS following a large break LOCA. It is also used to monitor LPI flow during piggy back operation following a small break LOCA. The low pressure injection flow to the reactor (2 channels, one associated with each LPI injection line) is determined from a differential pressure measurement calibrated to a span of 0 gpm to 5000 gpm.

The LPI flow indication is used by the operator to throttle the flow to ≤ 2000 gpm prior to switching the pump suction from the BWST to the RB sump. This assures adequate net positive suction head (NPSH) is maintained to the pump. The indication is also used to verify LPI flow to the reactor as a prerequisite to termination of HPI flow.

Since low pressure injection flow is a Type A variable on which the operator bases manual actions required for event mitigation for which no automatic controls are provided, it has been included in this LCO.

21. Degrees of Subcooling

Two channels of subcooling margin with inputs from RCS hot leg temperature (T_H), core exit temperature, and RCS pressure are provided. Multiple core exit temperatures are auctioneered with only the highest temperature being input to the monitor. A note has been added to indicate that the two channels of subcooling margin are backed up by either of two indications of subcooling margin based on similar inputs through the Safety Parameter Display System (SPDS). At least one SPDS channel must be available to provide this backup. With both SPDS channels INOPERABLE, Condition C is applicable. This is considered necessary because the core exit thermocouple inputs to the subcooling margin monitors are not environmentally qualified. The T_H inputs to the subcooling margin monitors and SPDS operate over a range of 120 to 920°F. The core exit temperature inputs operate over a range of 150 to 2000°F and 150

(continued)

BASES

LCO

21. Degrees of Subcooling (continued)

to 2500°F for the subcooling margin monitors and SPDS, respectively. RCS pressure inputs operate over a range of 200 to 2500 psig.

The subcooling margin monitors are used to verify the existence of, or to take actions to ensure the restoration of subcooling margin. Specifically, a loss of adequate subcooling margin during a LOCA requires the operator to trip the reactor coolant pumps (RCP's), to ensure high or low pressure injection, and raise the steam generator levels to the inadequate subcooling margin level. Since degrees of subcooling is a Type A variable on which the operator bases manual actions required for event mitigation for which no automatic controls are provided, it has been included in this LCO.

22. Emergency Diesel Generator, kW Indication

The Emergency Diesel Generator (EDG) provides standby (emergency) electrical power in the case of Loss of Offsite Power (LOOP). EDG kW indication is provided in the control room to monitor the operational status of the EDG.

EDG Power (kW) output indication is a type A variable because EDG kW indication provides the control room operator EDG load management capabilities. EDG load management enables the operator to base manual actions of load start and stop for event mitigation.

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING TECHNICAL SPECIFICATION CHANGES REGARDING

THE POST-ACCIDENT MONITORING SYSTEM

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated July 29, 1997, as supplemented by letter dated October 29, 1997, the Florida Power Corporation (FPC) proposed a revision to Crystal River 3 (CR-3) Technical Specifications (TS) to: (a) add subcooling margin monitors; (b) add decay heat removal (low pressure injection) flow to the post-accident monitoring (PAM) instrumentation; (c) add emergency diesel generator (EDG) kilowatt indication to the PAM instrumentation to support the CR-3 restart issue of emergency diesel generator load management; (d) revise the required thermocouple distribution in the assurance of availability of temperatures across the core; and (e) revise the description of containment pressure indication for post-accident monitoring. The clarifying information provided in the October 29, 1997, letter did not affect the original no significant hazards determination.

2.0 EVALUATION

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Events. The OPERABILITY of PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.

2.1 Subcooling Margin Monitors

FPC conducted a review of the plant Emergency Operating Procedures (EOPs) and determined that the "degrees of subcooling" instrumentation in the EOPs monitors were a Type A variable. Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled

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Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," specifies that instrumentation that monitors Type A variables should be designed to Category 1 criteria. Type A variables are those that provide primary information needed to permit the control room personnel to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis events. This instrumentation is considered to be a Type A variable since degrees of subcooling are used as criteria for manual initiation of high pressure injection, tripping of reactor coolant pumps and selection of the steam generator high level setpoint on the Emergency Feedwater Initiation and Control System during loss-of-coolant accidents (LOCAs).

The subcooling margin monitoring instrumentation at CR-3 is comprised of two trains of instrumentation. Each train has the following inputs: two reactor coolant system (RCS) hot leg temperature signals; two RCS pressure wide range signals; two RCS pressure narrow range signals; and eight incore thermocouple signals.

Two of the hot leg temperature signals, one for each monitor, originate at RC-4A-TE1 (Train A) and RC-4B-TE4 (Train B) and go to separate remote shutdown auxiliary cabinets in the respective 4160 volt engineered safeguards switchgear room. From the switchgear room, this instrumentation goes into the non-nuclear instrumentation (NNI) cabinets and to the Safety Parameter Display System (SPDS) in the main control room.

The other two hot leg temperature signals, one for each monitor, originate at RC-4B-TE1 (Train A) and RC-4A-TE4 (Train B) and go directly to separate NNI cabinets and from there to the SPDS in the main control room.

The two wide range RCS pressure signals that feed both trains of the subcooling margin monitor originate at RC-3A-PT3 and RC-3B-PT3 and go to separate engineered safeguards cabinets and the SPDS where they are used in subcooling margin calculations when RCS pressure is greater than 600 psig.

The "A" side narrow range RCS pressure signal feeds both trains of the subcooling margin monitor. It originates at RC-147-PT and goes to the "A" remote shutdown auxiliary cabinet in the "A" 4160 volt engineered safeguards switchgear room. From the switchgear room, it goes to the SPDS where it is used in subcooling margin calculations when RCS pressure is below 600 psig.

As part of the enhancement to the subcooling margin instrumentation feeding the SPDS, the licensee intends to provide a redundant "B" side narrow range pressure signal to both trains of the subcooling margin monitor. This narrow range RCS pressure signal will originate at RC-148-PT, and go to the "B" remote shutdown auxiliary cabinet in the "B" 4160 volt engineered safeguards switchgear room. From the switchgear room, it will go to the SPDS where it will be used in subcooling margin calculations when RCS pressure is below 600 psig.

The 16 incore thermocouple signals go into the Reactor Coolant Inventory and Tracking System (RCITS) cabinet in the control complex, with 8 of the 16 signals going to one subcooling margin monitor and the other 8 going to the other monitor. From the RCITS cabinets, the signals go to SPDS, 8 to each train, where they are auctioneered in the software so that the highest reading incore temperature is used in the subcooling margin calculation.

The staff agrees with the licensee's classification and considers that the subcooling margin monitors would provide adequate indications to the operators under post-accident conditions. Therefore, the proposed TS changes are acceptable.

In addition, the staff notes that the licensee intends to implement the following enhancements to the subcooling monitors: addition of a redundant "B" side narrow range RCS pressure signal; the enhancement to the seismic design of the existing SPDS system components; and the separation of trains and power from safety-related invertors which are backed up by the station standby power sources (EDGs).

2.2 Low Pressure Injection Flow

The licensee has classified low pressure injection flow as a Type A variable since low pressure injection flow must be manually throttled prior to switching from the borated water storage tank to the containment sump in order to prevent loss of net positive suction head. This instrumentation meets the RG 1.97 Category 1 criteria of redundant divisions and each has a single qualified channel of flow instrumentation. Therefore, the licensee's low pressure injection system flow instrumentation, and the proposed TS for this instrumentation, is acceptable.

2.3 EDG Kilowatt Indication

The licensee proposed adding EDG kilowatt indication to the post-accident-sampling-system (PASS) LCOs since it had recently been reclassified as Type A per RG 1.97. The indication would enable the operator to monitor load and perform load management (determine if additional loads could be added to the EDG under post-accident conditions) on the EDGs during post-LOCA conditions. The licensee indicated that instrumentation used for this function will meet the design criteria for RG 1.97 Category 1 instruments prior to restart from the current outage. The staff agrees with the reclassification; therefore, the proposed TS change is acceptable.

2.4 Thermocouple Distribution

The proposed change would require three instruments per core quadrant for the core exit temperature function as opposed to two sets of five for the entire core presently required by the TS. This change assures that a more representative distribution of temperatures across the core will be available to the operator. The revision presents a more logical relationship to the installed configuration than the existing requirement. The staff considers this change acceptable since three instruments per quadrant would provide better assurance that a representative distribution of temperatures across the core will be available to the operators under PAM conditions.

2.5 Containment Pressure

The licensee proposed to revise the description of the Containment Pressure instrumentation for PAM from "Narrow Range" to "Expected Post-Accident Range." The descriptor was proposed to eliminate confusion with the instruments used to monitor containment pressure during normal operation which have a range narrower (-5 to +5 psig) than the instruments controlled by the PAM TS (-10 to +70). No plant changes are being made in association with this request since the only change is in the description of the instruments which are currently installed for PAM. The staff agrees with this editorial change and considers the change acceptable.

3.0 STATE CONSULTATION

Based upon written notice of the proposed amendment, the Florida State official had no comments.

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,~~or are administrative.~~ 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding (62 FR 43369) . Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) ~~and (10)~~. 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

Based on the discussion is section 2, the staff concludes that the proposed changes to the PAM system such as the additions of subcooling margin monitors, decay heat removal flow, emergency diesel generator kilowatt indication, and the assurance of better indication across the core to the operators are consistent with the recommendations of RG 1.97, and are, therefore, acceptable. The revised description of the Containment Pressure instrumentation for PAM is considered an editorial change and is acceptable.

Principal Contributor: F. Gee

Date: December 22, 1997