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

January 22, 1993

Docket No. 50-302

Mr. Percy M. Beard, Jr.
Senior Vice President,
Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear
Operations Licensing
P.O. Box 219-NA-2I
Crystal River, Florida 34423-0219

Dear Mr. Beard:

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT RE: FIRE SERVICE SYSTEM (TAC NO. M79882)

The Commission has issued the enclosed Amendment No.147 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TS) in response to your application dated February 22, 1991 as modified by letters dated November 6, 1991, and May 28, 1992, and clarified by your letter dated December 21, 1992.

This amendment removes certain fire protection items from the CR-3 TS, modifies a license condition to establish the change control process for the Fire Protection Program, and corrects the remote shutdown system steam generator level instrumentation location and measurement range to reflect the existing plant configuration. The requirements of the fire protection TS have been relocated to the Fire Protection Plan.

Discussions with your staff, confirmed by your letter dated December 21, 1992, indicate that the reactor coolant temperature measurement range shown in the submittal of November 6, 1991 as unchanged, should in fact be revised in accordance with the February 22, 1991 change request.

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

Sincerely,

(Original Signed By Jan Norris For)

Harley Silver, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures
1. Amendment No.147 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PDII-2	PM:PDII-2	PD:PDII-2	SPLB	OGC
NAME	ETana ETT	HSilver	HBerkow	CMcCracken	S Hom w/changes
DATE	12/23/92	12/23/92	12/24/92	~~12~~/1/~~92~~ 93	~~12~~/14/~~92~~ 1/14/93

OFFICIAL RECORD COPY
FILENAME: CR79882.LTR

Mr. Percy M. Beard
Florida Power Corporation

Crystal River Unit No.3
Generating Plant

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DATED January 22, 1993

AMENDMENT NO. 147 TO FACILITY OPERATING LICENSEE NO. DPR-72-CRYSTAL RIVER UNIT 3

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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
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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. 147
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 February 22, 1991, as supplemented November 6, 1991, May 28, 1992, and December 21, 1992, 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. 147, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

In addition, paragraph 2.C.(9) of Facility License No. DPR-72 is amended to read as follows:

Fire Protection

Florida Power Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports dated July 27, 1979, January 22, 1981, January 6, 1983, July 18, 1985 and March 16, 1988, subject to the following provisions:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 22, 1993

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

ATTACHMENT TO LICENSE AMENDMENT NO. 147

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 areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
IV	IV
VII	VII
XII	XII
3/4 3-35	3/4 3-35
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 7-38	3/4 7-38
3/4 7-39	3/4 7-39
3/4 7-40	3/4 7-40
3/4 7-41	3/4 7-41
3/4 7-42	3/4 7-42
3/4 7-43	3/4 7-43
3/4 7-44	3/4 7-44
3/4 7-45	3/4 7-45
3/4 7-46	3/4 7-46
3/4 7-47	3/4 7-47
6-1	6-1
6-1a	6-1a
6-5	6-5
6-17	6-17
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 7-6	B 3/4 7-6

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION

BASES

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

	INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1.	Reactor Trip Breaker Indication	CRD switchgear room 124 foot elevation	open-close	1 per trip breaker an 1 per secondary trip breake
2.	Reactor Coolant Temperature - Th	Remote shutdown panel	120-920°F	1 per loop
3.	Reactor Coolant Pressure	Remote shutdown panel	0-2500 psig	1
4.	Pressurizer Level	Remote shutdown panel	0-320" H_2O	1
5.	Steam Generator Pressure	Remote shutdown panel	0-1200 psig	1 per steam generator
6.	Steam Generator Level	Remote shutdown panel	0-100%	1 per steam generator
7.	Decay Heat Removal Temperature	Remote shutdown panel	0-300°F	1 per cooler
8.	Motor Driven Emergency Feedwater Pressure	Intermediate Building 95 foot elevation	0-2000 psig	1 per pump
9.	Nuclear Services Closed Cycle Cooling Pumps Discharge Pressure	Auxiliary Building 95 foot elevation	0-300 psig	1
10.	Nuclear Services Closed Cycle Cooling Cooler Outlet Temperature	Auxiliary Building 95 foot elevation	0-250°F	1 per cooler

TABLE 4.3.6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N.A.
2. Reactor Coolant Temperature-Th	M	R(1)
3. Reactor Coolant Pressure	M	R
4. Pressurizer Level	M	R
5. Steam Generator Level	M	R
6. Steam Generator Pressure	M	R
7. Decay Heat Removal Temperature	M	R
8. Motor Driven Emergency Feedwater Pressure	M	R
9. Nuclear Services Closed Cycle Cooling Pumps Discharge Pressure	M	R
10. Nuclear Services Closed Cycle Cooling Cooler Outlet Temperature	M	R

(1) The CHANNEL CALIBRATION requirement for this function does not have to be met until the end of the cycle 8 refueling outage.

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Pages 3/4 3-42 thru 3/4 3-52 are deleted.

Next page is 3/4 3-53

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3) and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.10.1.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

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3/4.7.13 RADIOACTIVE WASTE SYSTEMS

WASTE GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.7.13.1 The quantity of radioactivity contained in each Waste Gas Decay Tank shall be limited to less than or equal to 39000 curies (considered as Xe 133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactivity in any Waste Gas Decay Tank exceeding the above limit, immediately suspend all additions of radioactive material to that tank, and within 48 hours reduce the tank contents to within its limit.

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

SURVEILLANCE REQUIREMENTS

4.7.13.1 The quantity of radioactive material contained in each Waste Gas Decay Tank shall be determined to be within the limit at least once per 7 days whenever radioactive materials are being added to the tank, and at least once per 24 hours during primary coolant system degassing operations.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Director, Nuclear Plant Operations shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for facility operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communications shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Final Safety Analysis Report.

b. The Director, Nuclear Plant Operations shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President, Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 FACILITY STAFF

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

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

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. DELETED

g. The shift supervisors shall hold a senior reactor operator license. The operators required to have a license shall hold at least a reactor operator license.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Chemistry and Radiation Protection Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Operations Technical Advisor, who shall have a Bachelor's degree, or the equivalent, in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Vice President, Nuclear Operations and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW COMMITTEE (PRC)

FUNCTION

6.5.1.1 The Plant Review Committee shall function to advise the Director, Nuclear Plant Operations (DNPO) on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Review Committee shall be composed of nine members and one chairman from Nuclear Operations Supervisory Personnel responsible for the following areas: operations, health physics, security, maintenance, modifications, engineering, nuclear safety, and quality. These positions will be designated by the DNPO in Administrative Procedures.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRC Chairman to serve on a temporary basis; no more than two alternates shall participate as voting members in PRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRC shall meet at least once per calendar month and as convened by the PRC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PRC shall consist of the Chairman or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Committee shall be responsible for:

a. Review of 1) all procedures and changes thereto as required by Specification 6.8.2, 2) and other proposed procedures or changes thereto as determined by the Director, Nuclear Plant Operations to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety, and changes to radwaste systems which could significantly alter their ability to meet Appendix I.

e. Investigation of all violations of the Technical Specifications including the review and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President, Nuclear Operations and to the Chairman of the Nuclear General Review Committee.

f. Review of all REPORTABLE EVENTS.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Chairman of the Nuclear General Review Committee.

i. Review of the Plant Security Plan and implementing procedures.

j. Review of the Emergency Plan and implementing procedures.

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SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below. A separate Licensee Event Report, when required by 10 CFR 50.73(a), need not be submitted if the Special Report meets the requirements of 10 CFR 50.73(b) in addition to the requirements of the applicable referenced Specification.

a. ECCS Actuation, Specification 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Deleted.

f. Specific Activity, Specification 3.4.8.

g. Results of Steam Generator Tube Inspection. Specification 4.4.5.5.b.

h. Deleted.

i. Deleted.

j. Deleted.

k. Deleted.

l. Deleted.

m. Deleted.

n. Deleted.

o. Deleted.

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM (RPS) AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

The OPERABILITY of the RPS and ESFAS instrumentation systems ensure that 1) the associated ESFAS action and/or RPS trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS and ESFAS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overal system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The ESFAS Functional Unit CHANNEL FUNCTIONAL TESTS shall be performed in accordance with Regulatory Guide 1.22 (Revision 0, 1972).

The measurement of response time at the specified frequencies provides assurance that the RPS and ESFAS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with reponse times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such test demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

The CHANNEL CALIBRATION of the Reactor Building High Radiation Monitor is performed in situ for at least one decade below 10 rad/hr. In situ calibration by electronic signal substitution is used for all range decades above 10 rad/hr.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Bases Figures 3-1 and 3-2 for examples of acceptable minimum incore detector arrangements.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event so that the response of those features important to safety may be evaluated. This capability is required to permit comparison of the measured response to that used in the design basis for the facility. This instrumentation is consistent with the recommendations of Safety Guide 12 "Instrumentation for Earthquakes", March 1971.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the needs for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs", February 1972.

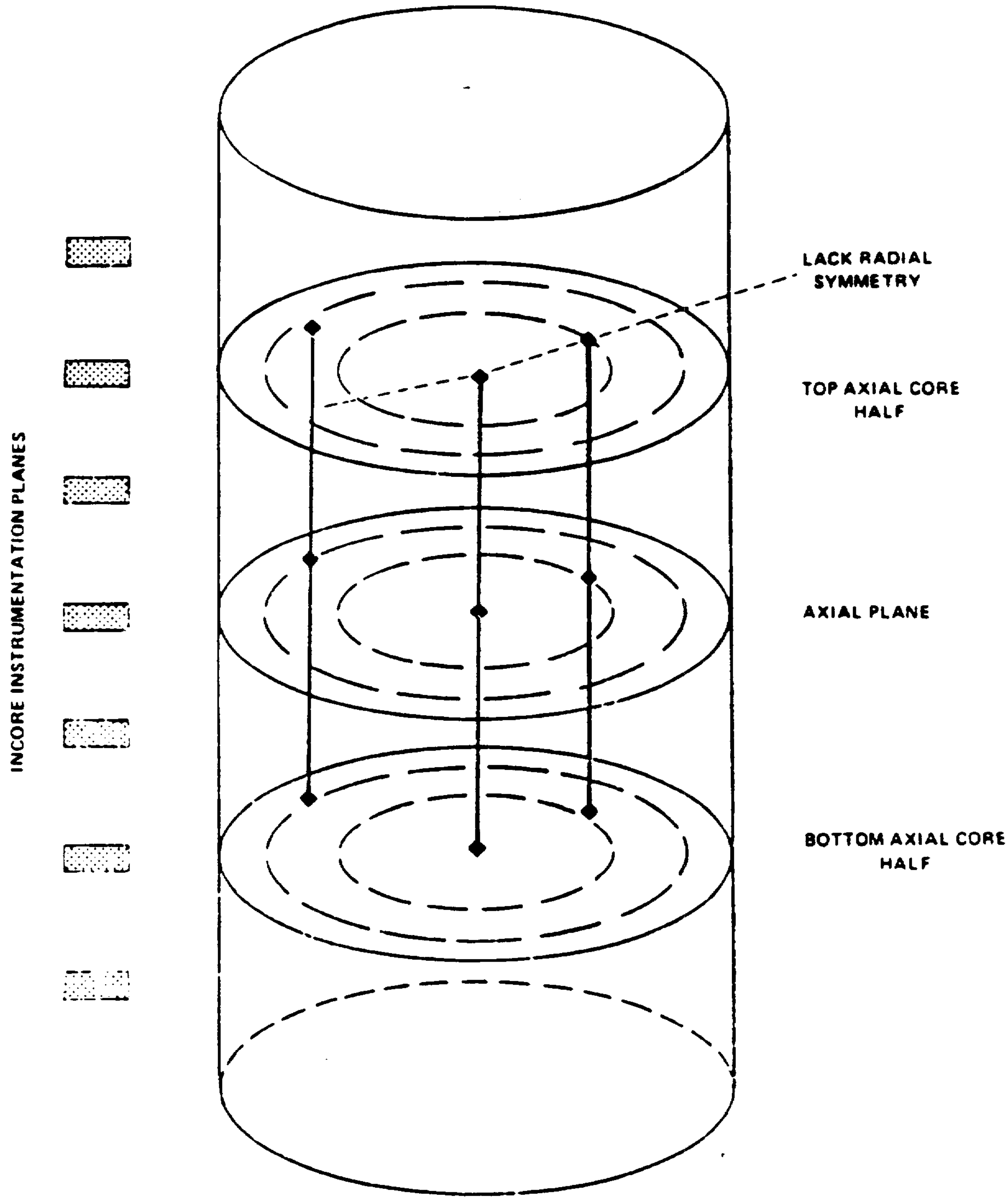
3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room capability is lost and is consistent with General Design Criterion 19 of Appendix "A", 10 CFR 50.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident", December 1975.

3/4.3.3.7 DELETED



Bases Figure 3-1 Incore Instrumentation Specification
Acceptable Minimum AXIAL POWER IMBALANCE Arrangement

1

3/4.7.8 AUXILIARY BUILDING VENTILATION EXHAUST SYSTEM

The OPERABILITY of the Auxiliary Building ventilation exhaust system ensures that suitable ambient conditions for personnel and equipment are maintained for all operating periods and that the effects of post accident conditions in the Auxiliary Building are mitigated. Supply and exhaust duct systems are arranged to direct air from areas of low to higher activity eventually directing it to the main exhaust filter system and from there through the fans to the exhaust vent. The main exhaust filters include roughing, HEPA, and charcoal cells.

3/4.7.9 HYDRAULIC SNUBBERS

The hydraulic snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. The only snubbers excluded from this inspection program are those installed on nonsafety-related systems, and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The inspection frequency applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Observed failures of these sample snubbers will require functional testing of additional units. To minimize personnel exposures, snubbers installed in high radiation zones or in especially difficult to remove locations may be exempted from these functional testing requirements provided the OPERABILITY of these snubbers was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.11 DELETED

3/4.7.12 DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated February 22, 1991, as modified by letters dated November 6, 1991 and May 28, 1992, and subsequently clarified by letter dated December 21, 1992, Florida Power Corporation (the licensee) submitted a request for changes to the Crystal River Unit 3 (CR-3) Technical Specifications (TS). The requested changes would delete the fire protection TS and their associated bases and definitions from the TS. The deleted requirements have been relocated to the CR-3 Fire Protection Plan. Existing TS require: (1) that written procedures be established, implemented and maintained for activities involving implementation of the Fire Protection Program, (2) periodic review of the Fire Protection Program and implementing procedures by a qualified individual/organization, and (3) review of changes to the Fire Protection Program and implementing procedures by the Plant Review Committee. License Condition 2.C.(9) would be revised to: (1) require the licensee to implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report (updated) as approved in the Fire Protection Safety Evaluation Reports dated July 27, 1979, January 22, 1981, January 6, 1983, July 18, 1985 and March 16, 1988, and (2) permit the licensee to make changes to the approved Fire Protection Program without prior approval of the NRC only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. The proposed changes are in accordance with the guidance provided in NRC Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications," dated August 2, 1988, and NRC Generic Letter 86-10, "Implementation of Fire Protection Requirements," dated April 24, 1986.

In addition, the steam generator level instrumentation location and measurement range, and the reactor coolant temperature instrument range in TS Remote Shutdown Monitoring Instrumentation Table 3.3-9 have been corrected to reflect the existing configuration of the plant.

2.0 BACKGROUND

Following the fire at the Browns Ferry Nuclear Power Plant on March 22, 1975, the Commission undertook a number of actions to ensure that improvements were implemented in the Fire Protection Programs for all power reactor facilities. Because of the extensive modification of Fire Protection Programs and the number of open issues resulting from staff evaluations, a number of revisions and alterations occurred in these programs over the years. Consequently, licensees were requested by Generic Letter 86-10 to incorporate the final NRC-approved Fire Protection Program in their Final Safety Analysis Reports (FSARs). In this manner, the Fire Protection Program including the systems, the administrative and technical controls, the organizations, and other plant features associated with fire protection would have a status consistent with that of other plant features described in the FSAR. In addition, the Commission concluded that a standard license condition, requiring compliance with the provisions of the Fire Protection Program as described in the FSAR, should be used to ensure uniform enforcement of fire protection requirements. Finally, the Commission stated in GL 86-10 that with the requested actions, licensees may request an amendment to delete the fire protection TS that would now be unnecessary.

The licensees for the Callaway and Wolf Creek plants submitted lead-plant proposals to remove fire protection requirements from their TS. This action was an industry effort to obtain NRC guidance on an acceptable format for license amendment requests to remove fire protection requirements from TS. Additionally, in the licensing review of new plants, the staff has approved applicant requests to remove fire protection requirements from TS issued with the operating license. Thus, on the basis of the lead-plant proposals and the staff's experience with TS for new licenses, Generic Letter 88-12 was issued to provide guidance on removing fire protection requirements from TS.

3.0 EVALUATION

Generic Letter 86-10 recommended the removal of fire protection requirements from the TS. Although a comprehensive Fire Protection Program is essential to plant safety, the basis for this recommendation is that many details of this program that are currently addressed in TS can be modified without affecting nuclear safety. Such modifications can be made, provided that there are suitable administrative controls over these changes. These details, that are presently included in TS and which are removed by this amendment, do not constitute performance requirements necessary to ensure safe operation of the facility and, therefore, do not warrant being included in TS. At the same time, suitable administrative controls ensure that there will be careful review and analysis by competent individuals of any changes in the Fire Protection Program including those technical and administrative requirements removed from the TS to ensure that nuclear safety is not adversely affected. These controls include: (1) the TS administrative controls that are applicable to the Fire Protection Program; (2) the license condition on implementation of, and subsequent changes to the Fire Protection Program, and (3) the 10 CFR 50.59 criteria for evaluating changes to the Fire Protection Program as described in the FSAR. The specific details relating to fire protection requirements removed from TS by this amendment include those specifications for detection systems, fire suppression systems, fire barriers, and fire brigade staffing requirements. The administrative control requirements include Fire Protection Program implementation as an element for which written procedures must be established, implemented and maintained. In addition, the technical review responsibilities of the Director, Nuclear Operations and the Plant Review Committee include the review of the Fire Protection Program and implementation procedures and submittal of recommended changes.

The TS changes proposed by the licensee in accordance with the guidance provided by Generic Letter 88-12 are as follows:

1) The requirements of TS 3/4.3.3.7 (Fire Detection Instrumentation), 3/4.7.11.1 (Fire Suppression Water System), 3/4.7.11.2 (Deluge and Sprinkler System), 3/4.7.11.3 (Halon System), and 3/4.7.12 (Fire Barrier Penetrations) would be relocated to the Fire Protection Plan and the FSAR.

2) The requirements of TS 6.1.2 and 6.1.3 (responsibility for the fire protection program) would be incorporated in the Fire Protection Plan and the FSAR.

3) The requirement of TS 6.2.2(f) (plant staff fire brigade composition) would be relocated to the Fire Protection Plan and referenced in the FSAR.

4) The requirements of TS 6.4.2 (fire brigade training) would be relocated to the Fire Protection Plan and the FSAR.

5) The requirements of TS 6.9.2(e) and (h) to submit the special reports to NRC will be removed from the TS.

The licensee proposed changing license condition 2.C.(9) to read as follows:

> Florida Power Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports dated January 27, 1979, January 22, 1981, January 6, 1983, July 18, 1985 and March 16, 1988, subject to the following provisions:
>
> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The licensee confirmed that the operational conditions, remedial actions, and test requirements associated with the removed fire protection TS will be incorporated in the Fire Protection Plan. This is in accordance with guidance of Generic Letter 88-12.

The licensee has stated in the February 22, 1991 submittal that the existing TS for the Remote Shutdown Panel (RSP) will be maintained in the TS and does not propose the removal of the RSP from the TS. This is consistent with the Generic Letter 88-12 guidance and is therefore acceptable.

With regard to the changes to TS Table 3.3-9, the changes are consistent with our previous acceptance of the remote shutdown capability of CR-3 and reflect the actual plant configuration.

Based on our review of the licensee's request for changes to the license and fire protection portions of the TS for CR-3, we conclude that FPC has followed the guidance provided by the NRC in Generic Letters 86-10 and 88-12, which provide adequate protection for public health and safety, and that the requested changes are acceptable.

Based on our review of the licensee's submittals, we conclude that the changes to TS Table 3.3-9 are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has 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 (56 FR 37583). 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.

6.0 CONCLUSION

The Commission has 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, (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: January 22, 1993