Template NRR-058

March 16, 2000

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

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING CONTAINMENT TENDON SURVEILLANCE PROGRAM (TAC NO. MA4966)

Dear Mr. Cowan:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 191 to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). This amendment is in response to a Florida Power Corporation (FPC) request dated February 19, 1999, as supplemented on February 23, 2000. FPC proposed changes to the CR-3 Improved Technical Specifications related to the Containment Tendon Surveillance Program. The proposed changes were a result of revisions to Title 10, *Code of Federal Regulations* (10 CFR) Section 50.55a

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,

/RA/

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.191to 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

March 16, 2000

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L. Wiens, Senior Project Manager, Section 2
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cc w/enclosures: See next page

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.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
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 19, 1999, as supplemented February 23, 2000, 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. 191, 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

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: March 16, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 191

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 contain vertical lines indicating the area of change.

Remove Page	Insert Page
3.6-2	3.6-2
5.0-11	5.0-11
5.0-29	5.0-29
B 3.6-4	B 3.6-4
B 3.6-5	B 3.6-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program. The maximum allowable leakage rate, L_a, is 0.25% of containment air weight per day at the calculated peak containment pressure, P_a.	In accordance with the Containment Leakage Rate Testing Program.
SR 3.6.1.2	Verify containment structural integrity in accordance with ITS 5.6.2.8.	In accordance with the Containment Inspection Program

5.6 Procedures, Programs and Manuals

5.6.2.6 Post Accident Sampling (continued)

c. Provisions for maintenance of sampling and analysis equipment.

5.6.2.7 Not used

5.6.2.8 Inservice Inspection Program

This program provides controls for inservice inspection of ASME Code Class 1, 2, 3, MC, and CC components, including applicable supports. The program shall include the following:

a. Provisions that inservice inspection of ASME Code Class 1, 2, 3, MC, and CC components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a;

b. The provisions of SR 3.0.2 are applicable to the frequencies for performing inservice inspection activities;

c. Inservice inspection of each reactor coolant pump flywheel shall be performed at least once every ten years. The inservice inspection shall be either an ultrasonic examination of the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination for exposed surfaces of the disassembled flywheels. The recommendations delineated in Regulatory Guide 1.14, Positions 3, 4 and 5 of Section C.4.b shall apply.

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.7 Reporting Requirements

5.7.2 Special Reports (continued)

The following Special Reports shall be submitted:

a. When a Special Report is required by Condition B or F of LCO 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

b. Any abnormal degradation of the containment structure found during the inspection performed in accordance with ITS 5.6.2.8 shall be reported to the NRC within 30 days of the current surveillance completion. The abnormal degradation shall be defined as findings such as delamination of the dome concrete, widespread corrosion of the liner plate, corrosion of prestressing elements (wires, strands, bars) or anchorage components extending to more than two tendons and group tendons force trends not meeting the requirements of 10CFR50.55a(b)(2)(ix)(B). The report shall include the description of degradation, operability determination, root cause determination and the corrective actions.

c. Following each inservice inspection of steam generator (OTSG) tubes, the NRC shall be notified of the following prior to ascension into MODE 4:

   1. Number of tubes plugged and repaired;

   2. Crack-like indications and assessment of growth for indications in the first span;

   3. Results of in-situ pressure testing, if performed; and

   4. Number of tubes and axially oriented TEC indications left in-service, the projected accident leakage, and an assessment of growth for TEC indications.

d. Results of OTSG tube inspections that fall into Category C-3 shall be reported to the NRC in accordance with 10CFR50.72.

e. The complete results of the OTSG tube inservice inspection shall be submitted to the NRC within 90 days after breaker closure following restart. The report shall include:

   1. Number and extent of tubes inspected,

   2. Location and percent of wall-thickness penetration for each indication of an imperfection,

   3. Location, bobbin coil amplitude, and axial and circumferential extent (if determined) for each first span IGA indication, and

   4. Identification of tubes plugged or repaired and specification of the repair methodology implemented for each tube.

BASES

ACTIONS

B.1 and B.2 (continued)

status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve with resilient seal leakage limits for SR 3.6.2.1 and 3.6.3.6 does not constitute a failure of this Surveillance unless the contribution from these penetrations causes overall Type A, B, and C leakage to exceed limits. SR Frequencies are as required by the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Inspection Program. Testing and Frequency are in accordance with Subsections IWE and IWL of the 1992 ASME Code and 10CFR 50.55a.

Abnormal degradation shall be determined by engineering evaluation. In the event abnormal degradation is detected, a Special Report shall be submitted in accordance with ITS 5.7.2.b. The impact of large-scale tendon degradation should also be evaluated with respect to Containment OPERABILITY. In this context, containment structural integrity is analogous to containment OPERABILITY.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B

2. FSAR, Sections 14.2.2

3. FSAR, 5.2.1.1

4. 1992 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL.

5. 10 CFR 100.

6. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J"

7. ANSI/ANS-56.8 1994, "American National Standard for Containment System Leakage Testing Requirement"

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE NO. DPR-72

CONTAINMENT TENDON SURVEILLANCE PROGRAM

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated February 19, 1999, as supplemented February 23, 2000, the licensee, Florida Power Corporation (FPC), requested an amendment to its facility Operating License No. DPR-72, for Crystal River Unit 3 (CR-3), in accordance with Title 10, *Code of Federal Regulations* (10 CFR) Section 50.90. The licensee proposed changes to the CR-3 Improved Technical Specification (ITS) Sections 3.6.1, 5.6.2.7, 5.6.2.8, and 5.7.2.b, related to the containment tendon surveillance program. The licensee stated that the proposed changes to the ITS resulted from revisions to 10 CFR 50.55a related to containment inspection. The revision to 10 CFR 50.55a became effective on September 9, 1996, and it required that the containment inspections be fully implemented by September 9, 2001.

The February 23, 2000, supplement did not affect the original proposed no significant hazards consideration determination, or expand the scope of the request as noticed in the *Federal Register* (64 FR 56530, October 20, 1999).

2.0 EVALUATION

The requirements in 10 CFR 50.55a were amended (61 FR 41303) to incorporate by reference Subsections IWE and IWL of the American Society of Mechanical Engineers Section XI Code (the Code) for inspection of containments of light water cooled reactors. Subsection IWE provides the requirements for inservice inspection, repair, and replacement of Class MC pressure retaining components (referred to as steel containments), and their integral attachments, and of metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments. Subsection IWL provides the requirements for preservice examination, inservice inspection and repair of the reinforced concrete and the post-tensioning systems of Class CC components (referred to as concrete containments). 10 CFR 50.55a (g)(6)(ii)(B) of the regulation requires that the first period inspection of containments be completed by September 9, 2001.

The following is a discussion of the specific sections of the ITS affected by the proposed amendment:

5.6.2.7 Containment Tendon Surveillance Program

The current ITS requires the licensee to perform the tendon surveillance in accordance with Regulatory Guide 1.35, Rev. 3; "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures, March 1990." The ITS also specifies that the provisions of Surveillance Requirements (SR) 3.0.2 and 3.0.3 regarding program inspection frequencies are applicable. The amendment proposes to delete Section 5.6.2.7 in its entirety from the ITS, since the tendon inspections will be covered by the Inservice Inspection Program and the requirements of 10 CFR 50.55a(b)(2)(ix).

Subsection IWL of the Code requires the inspection of containment post-tensioning tendons. These requirements, together with the additional requirements of 10 CFR 50.55a(b)(2)(ix), constitute acceptable requirements for the inspection of post-tensioning tendons in CR-3 containment. Moreover, the regulation requires the inspection of containment concrete (according to Subsection IWL), steel liner and penetrations (according to Subsection IWE), and potentially degraded inaccessible areas. Thus, the commitment to perform inspection as required by the regulation (in revised Section 5.6.2.8 of the ITS) ensures the integrity of the entire containment in addition to ensuring the integrity of the post-tensioning tendons as required by the current ITS. Also, the extent of deviation from the scheduled frequencies are satisfactorily covered in Subsections IWE and IWL of the Code. Thus, the staff finds the deletion of this section from the ITS acceptable.

5.6.2.8 Inservice Inspection Program

In this Section, the licensee proposes to incorporate the inservice inspection requirements of MC and CC components, including that of applicable supports, in accordance with the Code, as required by 10 CFR 50.55a. Section 5.6.2.8b of the current ITS requires that the provisions of SR 3.0.2 are applicable to the frequencies for performing inservice inspection activities related to the containment inspection. The staff finds the changes to this section of the ITS appropriately incorporate the requirements of 10 CFR 50.55a and are therefore acceptable.

5.7.2 Special Reports

Paragraph b. of the current Section 5.7.2 requires the licensee to report any abnormal degradation of the containment structure detected during the tests required by the Containment Tendon Surveillance Program within 30 days.

In the initial proposal, the licensee proposed to delete this requirement. However, after further discussion with the staff, the licensee revised its proposal to incorporate the following paragraph in lieu of the current requirement:

> Any abnormal degradation of the containment structure found during the inspection performed in accordance with ITS 5.6.2.8 shall be reported to the NRC within 30 days of the current surveillance completion. The abnormal degradation shall be defined as findings such as delamination of the dome concrete, widespread corrosion of the liner plate, corrosion of prestressing

elements (wires, strands, bars) or anchorage components extending to more than two tendons and group tendons force trends not meeting the requirements of 10 CFR 50.55a(b)(2)(ix)(B). The report shall include the description of degradation, operability determination, root cause determination and the corrective actions.

The incorporation of this paragraph in the ITS will ensure that the staff is alerted when the extent of containment degradation is significant. The staff finds the revision to this section of the ITS acceptable.

Additionally, the licensee proposed to revise Section 3.6.1.2 related to containment surveillance requirements to be consistent with the proposed revisions in Sections 5.6.2.7 and 5.6.2.8. This change maintains consistent wording of the ITS, and is 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. Nuclear Regulatory Commission (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. 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 (64 FR 56530). 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

Based on its review of the licensee's proposals, the staff has determined that the changes continue to provide reasonable assurance of containment integrity and thus 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.

Principal Contributor: H. Ashar, DE/EMEB/NRR

Date: March 16, 2000

Mr. John Paul Cowan
Florida Power Corporation

CRYSTAL RIVER UNIT NO. 3

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