April 8, 1999

Mr. John Paul Cowan
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing (SA2A)
Crystal River Energy Complex
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - STAFF EVALUATION AND ISSUANCE OF AMENDMENT REGARDING STEAM GENERATOR TUBE SURVEILLANCE PROGRAM (TAC NO. M99721)

Dear Mr. Cowan:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). By letter dated October 1, 1997, as supplemented on April 23 and November 17, 1998, and February 19, 1999, Florida Power Corporation submitted proposed changes to the Improved Technical Specifications (ITSs) for CR-3. The changes specify criteria for evaluating the growth of pit-like intergranular attack steam generator tube degradation identified in tubes in the "B" once-through steam generator (OTSG). FPC also requested to amend the ITSs to clarify the date by which the OTSG inservice inspection results are required to be submitted to the NRC.

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:

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

Docket No. 50-302

DFO1

Enclosures: 1. Amendment No.172 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

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Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING STEAM GENERATOR TUBE SURVEILLANCE PROGRAM (TAC NO. M99721)

Dear Mr. Cowan:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 172 to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). By letter dated October 1, 1997, as supplemented on April 23 and November 17, 1998, and February 19, 1999, Florida Power Corporation submitted proposed changes to the Improved Technical Specifications (ITSs) for CR-3. The changes specify criteria for evaluating the growth of pit-like intergranular attack steam generator tube degradation identified in tubes in the "B" once-through steam generator (OTSG). FPC also requested to amend the ITSs to clarify the date by which the OTSG inservice inspection results are required to be submitted to the NRC.

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,

Leonard A. Wiens, Senior Project Manager, Section 2
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Docket No. 50-302

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2. Safety Evaluation

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.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
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 October 1, 1997, as supplemented April 23 and November 17, 1998 and February 19, 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. 172, are hereby incorporated in the llicense. 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

Sheri R. Peterson

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

Date of Issuance: April 8, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 172

TO FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contains vertical lines indicating the area of change.

Remove Page	Insert Page
5.0-17	5.0-17
5.0-29	5.0-29

5.6 Procedures, Programs and Manuals

5.6.2.10 OTSG Tube Surveillance Program (continued)

8. Plugging/Sleeving Limit means the extent of degradation beyond which the tube shall be restored to serviceability by the installation of a sleeve or removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube or sleeve wall thickness. No more than five thousand sleeves may be installed in each OTSG.

9. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a main steam line or feedwater line break, as specified in 5.6.2.10.3.c, above.

10. Tube Inspection means an inspection of the entire OTSG tube as far as possible.

b. The OTSG shall be determined OPERABLE after completing the corresponding actions (plug or sleeve all tubes exceeding the plugging/sleeving limit and all tubes containing through-wall cracks) required by Table 5.6.2-2 (and Table 5.6.2-3 if the provisions of Specification 5.6.2.10.2.d are utilized). Defective tubes may be repaired in accordance with the B&W process (or method) equivalent to the method described in report BAW-2120P.

There are a number of OTSG tubes that have the potential to exceed the tube plugging/sleeving limit as a result of tube end anomalies. Defective tubes will be repaired or plugged during the next outage of sufficient duration. An evaluation has been performed which confirms that operability of the CR-3 OTSGs will not be impacted with those tube inservice.

c. Inservice tubes with pit-like IGA indications in the "B" OTSG first span shall be monitored for growth of these indications by using a test probe equivalent to the high frequency bobbin probe used in the 1997 inspection. The indicated percentage throughwall value from the current inspection shall be compared to the indicated percentage throughwall value from the 1997 inspection.

5.6.2.11 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control pints for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

(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 detected during the tests required by the Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

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 sleeved

   2. Crack-like indications in the first span

   3. An assessment of growth in the first span indications, and

   4. Results of in-situ pressure testing, if performed.

   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 and tubes sleeved.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated October 1, 1997, as supplemented on April 23 and November 17, 1998, and February 19, 1999, Florida Power Corporation (the licensee) submitted proposed changes to the Improved Technical Specifications (ITSs) for Crystal River Unit 3 (CR-3). The changes specify criteria for evaluating the growth of pit-like intergranular attack (IGA) steam generator tube degradation identified in tubes in the "B" once-through steam generator (OTSG). The licensee has also requested to amend the ITSs to clarify the date by which the OTSG inservice inspection results are required to be submitted to the U.S. Nuclear Regulatory Commission (NRC). The April 23 and November 17, 1998, and February 19, 1999 letters did not affect the original no significant hazards determination. The following documents the staff's assessment of the changes proposed by the licensee.

2.0 BACKGROUND

2.1 Regulatory Framework For Proposed Licensing Action

Steam generator tubing comprises a significant fraction of the reactor coolant pressure boundary. Title 10 of the Code of Federal Regulations (CFR), 10 CFR 50.55a(c) specifies that components that are part of the reactor coolant pressure boundary must be designed and constructed to meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. To ensure the continued integrity of the tubing at operating pressurized water reactor (PWR) facilities, 50.55a further requires that throughout the service life of a PWR facility, Class 1 components meet the requirements in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" of the ASME Code. This requirement includes the inspection and tube repair criteria of Section XI of the ASME Code. However, an exception is provided for design and access provisions and preservice examination requirements in Section XI. In addition, 10 CFR 50.55a(b)(2)(iii) states that if the technical specification (TS) surveillance requirements for steam generators differ from those in Article IWB-2000 of Section XI of the ASME Code, the inservice inspection program is governed by the TSs.

As part of the plant licensing basis, applicants for a PWR operating license analyze the consequences of postulated design basis accidents that assume degradation of the steam generator tubes such that primary coolant leaks to the secondary coolant side of the steam generators. Examples of such accidents are a steam generator tube rupture, a main steam line break, a locked rotor, and a control rod ejection. Analyses of these accidents consider the primary-to-secondary leakage that may occur during these postulated events when demonstrating that radiological consequences do not exceed the 10 CFR Part 100 guidelines, or some fraction thereof, for offsite doses, nor General Design Criterion 19 for control room operator doses. The staff uses criteria specified in NUREG-0800, the Standard Review Plan, to evaluate these accidents.

A plant's TSs require that licensees perform periodic inservice inspections of the steam generator tubing and repair or remove from service (by installing plugs in the tube ends) all tubes exceeding the tube repair limit. In addition, operational leakage limits are included in the TSs to ensure that, should tube leakage develop, the licensee will take prompt action to avoid rupture of the leaking tubes. These requirements are intended to ensure that burst margins are maintained consistent with Appendices A and B to 10 CFR Part 50 and that the potential for leakage is maintained consistent with what has been analyzed as part of the plant licensing basis.

Revision 1 of NRC Regulatory Guide (RG) 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," provides guidance concerning steam generator tube inspection scope and frequency and nondestructive examination methodology. RG 1.83 is referenced in the standard review plan and is intended to provide a basis for reviewing inservice inspection criteria in the TSs. NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," provides guidelines for determining the tube repair criteria and operational leakage limits that are specified in the TSs. Together, these two RGs provide specific criteria that should be considered in proposed steam generator tube alternate repair criteria in order to satisfy the previously mentioned regulatory requirements.

2.2 Description of Proposed Amendment

A number of tubes have been identified as containing apparent indications of pit-like IGA degradation in the "B" OTSG at CR-3 for a number of operating cycles. The licensee has attempted to develop alternate repair criteria to address this mode of degradation in previous cycles. However, the NRC has not accepted previous proposals developed by the licensee to disposition tubes with pit-like IGA indications on a permanent basis. In the 1997 inspections at CR-3, the licensee applied a method to estimate the depth of IGA degradation using traditional eddy current inspection techniques. The technique relied on using inspection data obtained with high frequency bobbin coil probes. Tubes with IGA flaws with depths measured in excess of the existing depth-based tube plugging limits in ITS 5.6.2.10.4.a.7 (i.e., 40 percent of nominal tube wall thickness) were removed from service. In order to ensure adequate steam generator tube integrity through the end of the operating cycle following the outage in which tubes are inspected, licensees typically complete operational assessments to conservatively estimate the conditional probability of tube rupture and steam line break primary-to-secondary leak rate through defective steam generator tubes. One critical factor in these assessments is the accuracy of growth rates for known degradation mechanisms.

The licensee has been inspecting tubes with pit-like IGA indications for a number of cycles. Assessments of IGA flaw growth rate completed by the licensee have concluded that this damage mechanism is dormant for flaws in the lower bundle region of the "B" OTSG. The staff has previously concluded that changes in the eddy current data acquisition equipment used in successive inspections may have hindered the licensee's ability to detect minor changes in flaw growth rate. Such changes could have altered the eddy current readings for flaws inspected during steam generator tube examinations. Therefore, the staff has not accepted the licensee's conclusion on pit-like IGA flaw progression. In addition, previous growth rate assessments were largely based on voltage change. The current approach used at CR-3 to disposition tubes with these indications is based on flaw depth. As such, overall growth rates will be determined on the basis of flaw depth rather than voltage in future inspections.

The licensee currently inspects the CR-3 steam generators using a mid-range frequency bobbin coil eddy current probe. Potential IGA indications detected during these inspections are reinspected using rotating probes to characterize the mode of degradation and a high frequency bobbin coil probe to estimate the through-wall depth of pit-like IGA degradation. During the 1997 steam generator tube examinations, the licensee established a baseline by which future degradation growth would be measured. In order to accurately assess any changes in the depth of degradation identified and measured in the baseline examination, subsequent examinations will use data acquisition equipment that is equivalent to the system used when the depth sizing technique was qualified prior to the 1997 inspections. The proposed ITS changes would establish this requirement for evaluating growth rates. Specifically, the growth of indications identified in future inspections will be determined by comparing the examination results to those obtained in the 1997 inspection using equivalent test probes.

The current CR-3 ITSs require the licensee to submit to the NRC the results of its OTSG inservice inspection within 90 days after completion of the inspections (ITS 5.7.2, "Special Reports"). The licensee has proposed to modify this requirement to mandate the submittal of this report within 90 days after restarting the unit (i.e., breaker closure). The purpose for this proposed change is to more clearly define the timing when the report is required to be sent to the NRC and to specify a requirement consistent with guidance issued by the NRC for submitting reports for generator tube alternate repair criteria.

3.0 STAFF EVALUATION

The current methodology for dispositioning tubes with pit-like IGA indications in the "B" OTSG at CR-3 relies on the determination of flaw through-wall depth using eddy current inspection techniques. In order to evaluate changes in the depth of previously identified degradation, the licensee will compare the depth determined in the most recent inspection to that determined from inspections completed in 1997. The resulting growth rate for an indication is the difference between the two measured indication depths. Indication depth measurements are obtained using eddy current probes equivalent to those used in the 1997 baseline examination. The requirement to use equivalent test probes will reduce errors in the calculated growth rate introduced by using different eddy current probes (e.g., mid-range bobbin probe) in two different inspections. Therefore, the staff concludes that the proposed ITS changes to TS 5.6.2.10.c, which provide that the licensee will determine flaw growth rate based on through-wall depth compared with the 1997 baseline examination are acceptable because data acquisition uncertainty is minimized, and the results can be directly related to the CR-3 steam generator tube plugging limit.

The existing ITSs for CR-3 state that the licensee shall submit complete results of the OTSG inservice inspection within 90 days following completion of the inspections. The licensee has proposed to modify this requirement in TS 5.7.2.c by stating that this report is due 90 days after unit startup from the outage (i.e., breaker closure). This change would slightly extend the date for submitting this report, but the revision would update the requirements at CR-3 to be consistent with those adopted by other PWR facilities. Any additional delay in submitting this report resulting from this change should not impact the NRC staff's ability to adequately assess any steam generator tube integrity issues stemming from the inspections in a reasonable period of time. In addition, other reporting requirements specified in the ITSs and 10 CFR 50.72 and 50.73 enable the NRC to stay informed of any identified conditions that could affect the health and safety of the public. Therefore, the staff concludes that the proposed change to specify that a complete inservice inspection report is due 90 days from restart is acceptable.

4.0 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.

5.0 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 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 (62 FR 54873). The amendment also changes reporting or record-keeping requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(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.

6.0 CONCLUSION

Based on its review of the licensee's proposal, the staff has determined that the proposed changes to the CR-3 ITSs will continue to provide adequate assurance of steam generator tube integrity because the growth rate of pit-like IGA degradation will be monitored in a manner that will minimize uncertainties introduced by the data acquisition equipment. In addition, the modification to the reporting requirements will not affect the prompt notification of the NRC of any potentially significant issues related to the inservice inspection of the OTSG tubing because alternate reporting requirements will ensure prompt notification of the staff of potentially safety significant issues. 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: Phillip J. Rush, DE/EMCB

Date: April 8, 1999

Mr. John Paul Cowan
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