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

May 29, 2009

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING THE REVISION OF THE STEAM GENERATOR PORTION OF THE TECHNICAL SPECIFICATIONS TO REFLECT THE REPLACEMENT OF THE STEAM GENERATORS (TAC No. MD9547)

Dear Mr. Young:

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 234 to Facility Operating License No. DPR-72 for Crystal River, Unit 3 (CR-3) in response to your letter dated August 28, 2008, as supplemented by letter dated January 19, 2009. The amendment revises the CR- 3 Technical Specifications (TSs) to implement the Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler, TSTF-449, Revision 4, "Steam Generator Tube Integrity," inspection requirements for the replacement once through steam generators (OTSGs) that are being installed during the CR-3 fall 2009 refueling outage. In support of the OTSG replacement, the amendment revises the CR-3 Facility Operating License and TSs 3.4.16, "Steam Generator (OTSG) Tube Integrity," 5.6.2.10, "Steam Generator (OTSG) Program," and 5.7.2, "Special Reports."

A copy of the safety evaluation is enclosed. The notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 234 to Facility Operating License DPR-72
2. Safety Evaluation

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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. 234
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 August 28, 2008, as supplemented by letter dated January 19, 2009, 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. 234 , 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 upon startup from Refueling Outage R16.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: May 29, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 234

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following page of Facility Operating License DPR-72 with the attached revised page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

Remove	Insert
4	4

Replace the following page of the Appendix "A" Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3.4-34	3.4-34
3.4-35	3.4-35
5.0-13	5.0-13
5.0-14	5.0-14
5.0-15	5.0-15
5.0-16	5.0-16
5.0-17	5.0-17
5.0-17A	--------
5.0-29	5.0-29

of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

2.C.(1) Maximum Power Level

Florida Power Corporation is authorized to operate the facility at a steady state reactor core power level not in excess of 2609 Megawatts (100 percent of rated core power level).

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 234 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

The Surveillance Requirements contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment 149. The Surveillance Requirements shall be successfully demonstrated prior to the time and condition specified below for each.

a) SR 3.3.8.2.b shall be successfully demonstrated prior to entering MODE 4 on the first plant start-up following Refuel Outage 9.

b) SR 3.3.11.2, Function 2, shall be successfully demonstrated no later than 31 days following the implementation date of the ITS.

c) SR 3.3.17.1, Functions 1, 2, 6, 10, 14, & 17 shall be successfully demonstrated no later than 31 days following the implementation date of the ITS.

d) SR 3.3.17.2, Function 10 shall be successfully demonstrated prior to entering MODE 3 on the first plant start-up following Refuel Outage 9.

e) SR 3.6.1.2 shall be successfully demonstrated prior to entering MODE 2 on the first plant start-up following Refuel Outage 9.

f) SR 3.7.12.2 shall be successfully demonstrated prior to entering MODE 2 on the first plant start-up following Refuel Outage 9.

g) SR 3.8.1.10 shall be successfully demonstrated prior to entering MODE 2 on the first plant start-up following Refuel Outage 9.

h) SR 3.8.3.3 shall be successfully demonstrated prior to entering MODE 4 on the first plant start-up following Refuel Outage 9.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Steam Generator (OTSG) Tube Integrity

LCO 3.4.16 OTSG tube integrity shall be maintained.

AND

All OTSG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each OTSG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more OTSG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or OTSG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or OTSG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR OTSG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify OTSG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.16.2	Verify that each inspected OTSG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a OTSG tube inspection

5.6 Procedures, Programs and Manuals

5.6.2.10 Steam Generator (OTSG) Program

A Steam Generator Program shall be established and implemented to ensure that OTSG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an OTSG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the OTSG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for OTSG tube integrity. OTSG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.10 OTSG Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than an OTSG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all OTSGs and leakage rate for an individual OTSG. Leakage is not to exceed one gallon per minute per OTSG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.12, "RCS Operational LEAKAGE."

c. Provisions for OTSG tube repair criteria. A tube found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for OTSG tube inspections. Periodic OTSG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that OTSG tube integrity is maintained until the next OTSG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each OTSG during the first refueling outage following OTSG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the OTSGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.10 OTSG Program (continued)

of the period and the remaining 50% by the refueling outage nearest the end of the period. No OTSG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any OTSG tube, then the next inspection for each OTSG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

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5.7 Reporting Requirements

5.7.2 Special Reports (continued)

5. Number of tubes plugged during the inspection outage for each active degradation mechanism,

6. Total number and percentage of tubes plugged to date,

7. The results of condition monitoring, including the results of tube pulls and in-situ testing.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated August 28, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082460317), as supplemented by letter dated January 19, 2009 (ADAMS Accession No. ML090360489), the Florida Power Corporation (the licensee) requested changes to the Technical Specifications (TSs) for Crystal River Unit 3 (CR-3). The proposed changes would revise the CR- 3 TSs to implement the Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler, TSTF-449, Revision 4, "Steam Generator Tube Integrity," inspection requirements for the replacement once through steam generators (OTSGs) that are being installed during the CR-3 fall 2009 refueling outage. The replacement OTSGs differ from the existing OTSGs in that the tube material is Alloy 690 thermally treated in the replacements versus Alloy 600 in the existing OTSGs. Additionally, this amendment removes inspection requirements that are designated for specific damage conditions in the existing OTSGs, remove tube repair techniques approved by the license amendment No. 233, dated May 16, 2007, for the existing OTSGs, and remove inspection and reporting requirements specific to those repair techniques. The amendment would revise the CR-3 Facility Operating License and TSs 3.4.16, "Steam Generator (OTSG) Tube Integrity," 5.6.2.10, "Steam Generator (OTSG) Program," and 5.7.2, "Special Reports," and is in support of the replacement of the OTSGs at CR-3 during the fall 2009 outage.

The supplement dated January 19, 2009, was included in the NRC staff's original proposed no significant hazards consideration determination, which was published in the *Federal Register* on February 24, 2009 (74 FR 8284).

2.0 REGULATORY EVALUATION

Steam generator tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the steam generator (SG) tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 states that the RCPB shall have "an extremely low probability of abnormal leakage ... and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards possible" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and leak tight integrity" (GDC 32). To this end, paragraph 50.55a of 10 CFR specifies that components that are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Paragraph 50.55a further requires, in part, that throughout the service life of a pressurized water reactor (PWR) facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional SG tube surveillance requirements in the TSs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture and main steamline break. These analyses consider the primary-to-secondary leakage through the tubing which may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 100 guidelines for offsite doses (or 10 CFR 50.67, as appropriate), GDC-19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis.

The CR-3 TSs are modeled after TSTF-449. Technical Specification 5.6.2.10 for CR-3 requires that an SG program be established and implemented to ensure that OTSG tube integrity is maintained. Tube integrity is maintained by meeting specified performance criteria for structural and leakage integrity consistent with the plant design and licensing bases. Technical Specification 5.6.2.10 requires a condition monitoring assessment be performed during each outage during which the OTSG tubes are inspected or plugged to confirm that the performance criteria are being met. Technical Specification 5.6.2.10 also includes provisions regarding the scope, frequency, and methods of OTSG tube inspections.

3.0 TECHNICAL EVALUATION

Crystal River Unit 3 currently has two Babcock and Wilcox OTSGs. Each SG contains 15,531 stress relieved mill-annealed Alloy 600 tubes. Each tube has a nominal outside diameter of 0.625 inches and a nominal wall thickness 0.034 inches. The tubes were mechanically roll expanded at both ends for approximately 1 inch of the 24-inch thick tubesheets and are supported by a number of carbon steel support plates.

The reactor coolant system operational leakage technical specification was modified in accordance with TSTF-449 by License Amendment No. 223 dated May 16, 2007. This was the approval of the application of the TSTF and CLIIP for the existing steam generators and remains in force for the replacement OTSGs.

The Nuclear Regulatory Commission (NRC or the Commission) has approved a few amendments related to the original CR-3 SGs. The licensee is permitted to repair tubes by sleeving and rerolling the portion of the tube in the tubesheet. In addition, the licensee is allowed to implement degradation specific repair criteria for tube end cracks and pit-like intergranular attack.

The replacement OTSGs differ from the existing OTSGs in that the tube material is thermally treated Alloy 690 in the replacement OTSGs versus the mill-annealed Alloy 600 in the existing SGs. The replacement OTSGs are scheduled to be installed during the fall 2009 refueling outage.

The licensee is proposing to remove the TSs requirements associated with alternate tube repair criteria applicable to their original OTSGs. These requirements are contained in TS 5.6.2.10.c (tube repair criteria), TS 5.6.2.10.d (tube inspection criteria), and TS 5.7.2 (reporting requirements). The licensee is also proposing to remove the TSs requirements associated with the tube repair methods. These requirements are contained in TS 3.4.16 (limiting condition for operation and associated action and surveillance requirements), TS 5.6.2.10.a (condition monitoring), TS 5.6.2.10.c (tube repair criteria), TS 5.6.2.10.d (tube inspections), TS 5.6.2.10.f (tube repair methods), and TS 5.7.2 (reporting requirements).

The staff finds these changes acceptable since these requirements were developed for the licensee's original OTSGs. With the planned replacement of the OTSGs, these requirements are no longer needed. In addition, given the design differences between the original and replacement OTSGs, these requirements are not applicable to the replacement OTSGs.

In addition, the licensee is proposing to adopt inspection requirements applicable to SGs with thermally treated Alloy 690 tubes, i.e., the material used in their replacement OTSGs. The staff finds these proposed changes acceptable since the licensee's replacement OTSGs have thermally treated Alloy 690 tubes rather than mill-annealed Alloy 600 tubes. The staff further finds that the proposed changes are consistent with TSTF-449. In their application, the licensee indicated that the inspection requirements in TSTF-449 for mill annealed Alloy 600 tubing and the thermally treated Alloy 690 tubing are expected to produce the same outcome, i.e., that tube integrity will continue to be maintained. This statement was made in the context of the prescriptive (or maximum) inspection intervals listed in the technical specifications. The staff agrees that the desired outcome of the inspections is the same regardless of the tube material and base this conclusion on the performance-based inspection requirements. The performance-based inspection requirements in the technical specifications require that inspection intervals be established so as to ensure that SG tube integrity is maintained until the next OTSG inspection. Due to OTSG design and operating/transient conditions, e.g., severe corrosive conditions, the prescriptive (or maximum) inspection interval requirements may not always ensure tube integrity. As such, the performance-based inspection requirement is essential for ensuring tube integrity.

In summary, the NRC staff finds that the proposed changes to the SG TSs requirements are acceptable since the resultant TSs are consistent with TSTF-449 and reflect the tube material in the replacement OTSGs.

4.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, NRC Senior Project Manager, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

The 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 NRC has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding issued on February 24, 2009 (74 FR 8284). 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.

6.0 CONCLUSION

The NRC 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Aloysius Obodoako

Date: May 29, 2009

May 29 2009

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING THE REVISION OF THE STEAM GENERATOR PORTION OF THE TECHNICAL SPECIFICATIONS TO REFLECT THE REPLACEMENT OF THE STEAM GENERATORS (TAC No. MD9547)

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A copy of the safety evaluation is enclosed. The notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/
Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 234 to Facility Operating License DPR-72
2. Safety Evaluation

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