July 27, 2005

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 REACTOR COOLANT PUMP FLYWHEEL SURVEILLANCE (TAC NO. MC4793)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 218 to Facility Operating License No. DPR-72 for Crystal River Unit 3. The amendment consists of changes to the existing Technical Specifications (TS) in response to your letter dated October 15, 2004.

The amendment revises surveillance requirements related to the reactor coolant pump flywheel inspections to extend the allowable inspection interval to 20 years.

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/
Brenda L. Mozafari, 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. 218 to DPR-72
2. Safety Evaluation

cc w/encls: See next page

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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. 218
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 15, 2004, 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:

    <u>Technical Specifications</u>

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 218, 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Michael L. Marshall, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 27, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 218

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following page of the Appendix "A" Technical Specifications 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
5.0-11	5.0-11

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 218 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 application dated October 15, 2004, Florida Power Corporation (the licensee, also doing business as Progress Energy Florida, Inc.) proposed changes to the Crystal River Unit 3 (CR-3) Technical Specifications (TS) (ADAMS Accession No. ML042950619). The requested change revises surveillance requirement Section 5.6.2.8 “Inservice Inspection Program” for CR-3 to extend the examination frequency for the reactor coolant pump (RCP) motor flywheel from the currently approved 10-year inspection interval to an interval not to exceed 20 years. This change is based on Technical Specification Task Force (TSTF) change traveler TSTF-421, “Revision to RCP Flywheel Inspection Program (WCAP-15666),” that has been approved generically for the Westinghouse Standard Technical Specifications (STS), NUREG-1431, related to the RCP flywheel inspections to extend the allowable inspection interval to 20 years. Although CR-3 is a Babcock and Wilcox plant not specifically addressed by the NRC Notice of Availability published on October 22, 2003 (68 FR 60422), the licensee states that its justification for incorporating TSTF-421 for CR-3 is based on the fact that its flywheels are bounded by the analyses performed in WCAP-15666, “Extension of Reactor Coolant Pump Motor Flywheel Examination.”

The NRC staff issued a model safety evaluation and model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the *Federal Register* on June 24, 2003 (68 FR 37590). The notice of availability of the model application was issued on October 22, 2003 (68 FR 60422). The licensee affirmed the applicability of the model NSHC determination in its application dated October 15, 2004.

2.0 REGULATORY EVALUATION

The function of the RCP in the reactor coolant system (RCS) of a pressurized-water reactor (PWR) plant is to maintain an adequate cooling flow rate by circulating a large volume of primary coolant water at high temperature and pressure through the RCS. Following an assumed loss of power to the RCP motor, the flywheel, in conjunction with the impeller and motor assembly, provides sufficient rotational inertia to assure adequate primary coolant flow during RCP coastdown, thus resulting in adequate core cooling.

A proposed justification for extending the RCP flywheel inspections from a 10-year inspection interval to an interval not to exceed 20 years was provided by the Westinghouse Owners Group (WOG) in Topical Report (TR) WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination," transmitted by letter dated August 24, 2001. The TR addressed the proposed extension for all domestic WOG plants. The NRC accepted the TR for referencing in license applications in a letter and safety evaluation (SE) dated May 5, 2003 (ADAMS Accession No. ML031250595).

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on October 22, 2003 (68 FR 60422), NRC Notice for Comment published on June 24, 2003 (68 FR 37590), TSTF-421, WCAP-15666, and the related NRC SE. CR-3 is a Babcock and Wilcox plant, not specifically addressed by the NRC Notice of Availability; however, the licensee stated that it is bounded by the analyses performed in WCAP-15666.

3.0 TECHNICAL EVALUATION

The WCAP-15666 risk evaluation assumptions that are used to define the envelope in order for individual licensees to confirm that the WCAP applies to them are identified below, together with CR-3 plant-specific data.

1. The probability of a large early release given an RCP motor flywheel failure is assumed to be 1.0 in WCAP-15666. This is a conservative and bounding assumption applicable to all PWRs including CR-3.

2. The conditional probability of loss of offsite power (LOOP) and consequential loss of power to the RCP given a loss-of-coolant accident (LOCA) and startup of the emergency core cooling system (ECCS) is estimated in NUREG/CR-6538 as 1.4E-2. The same value is conservatively used for LOOP following a general reactor trip (a general reactor trip places less demand on the electrical systems than the startup of the ECCS, and NUREG/CR-6538 estimates the conditional probability of a LOOP given a general transient reactor trip as about a factor of 10 lower). This generic conditional probability of 1.4E-2 for a conditional LOOP is used in WCAP-15666. CR-3 states that it has not experienced a LOOP due to a general reactor trip and that its design provides two separate switchyards for the main generator connection and the power supply to the safety busses. The NRC staff finds that use of the generic value for a conditional LOOP in a PWR obtained from NUREG/CR-6538 is reasonable and consistent with CR-3 operating experience.

3. The generic frequency for a general transient reactor trip is estimated as one (1.0) event per year in WCAP-15666. The current CR-3 probabilistic risk assessment model frequency for a reactor trip is slightly higher (i.e., 1.5 per year). The scenario of a reactor trip followed by a consequential LOOP is the dominant risk scenario for flywheel failure. Applying the plant-specific frequency for general transient reactor trip, the increase in large early release frequency (LERF) is from 1.2E-8 per year in WCAP-15666 to 3E-8 per year. This increase in LERF estimate remains significantly below the very small increase in the LERF guideline of 1E-7 per year (RG 1.174 guidance) when the plant-specific data is used.

4. The mean value for the frequency of large LOCA that is used in WCAP-15666 is 2E-06 per year. CR-3 states that its LOCA frequency estimate is 5E-6/yr, which is consistent with the generic PWR large LOCA frequency estimated in NUREG/CR-5750. As discussed in the staff SE on WCAP-15666, the large LOCA frequency estimate in WCAP-15666 is less than the estimate in NUREG/CR-5750 because only the failure of a limited amount of large diameter reactor coolant piping could cause the acceleration of the flywheel. The observation that the failure of only a limited amount of large-diameter reactor coolant piping could cause the acceleration of the flywheel also remains valid for CR-3. The combination of the frequency of a large LOCA, the conditional probability of a LOOP, and the probability of existence of a flaw that fails at the accelerated revolution per minute is a negligible contributor to increased risk in WCAP-15666, and the NRC staff finds that it remains negligible using the CR-3 LOCA estimates.

5. The material used for CR-3 flywheels are SA 516 and SA 508 pressure vessel quality vacuum improved steel plate that is estimated to have fracture toughness equivalent to that of SA 533, Grade B material analyzed in WCAP-15666.

6. From review of WCAP-14535A, “Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,” the following conclusions were noteworthy in regard to the analyses used to justify extension of the currently approved 10-year inspection interval to an interval not to exceed 20 years.

    The ductile failure limiting speed for CR-3 was determined to be higher than that of the flywheels analyzed in WCAP-15666.

    The flywheel deformation at overspeed conditions for CR-3 was estimated to be less than that of flywheels analyzed in WCAP-15666.

    Fatigue crack growth in CR-3 flywheels assuming 6000 startups and shutdowns was estimated to be less than that of the flywheels analyzed in WCAP-15666.

    Critical crack length for flywheel overspeed to 1500 RPM was greater for CR-3 flywheels than that of the flywheels analyzed in WCAP-15666 indicating greater flaw tolerance.

The NRC staff, therefore, has determined that CR-3 flywheels are bounded by the deterministic integrity evaluations in WCAP-14535A and the risk assessment performed in WCAP-15666.

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, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a surveillance requirement. 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 9992). 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 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.

Principal Contributors: P. Patnaik
S. Dinsmore

Date: July 27, 2005

cc:

Mr. R. Alexander Glenn
Associate General Counsel (MAC-BT15A)
Florida Power Corporation
P.O. Box 14042
St. Petersburg, Florida 33733-4042

Mr. Jon A. Franke
Plant General Manager
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Mr. Jim Mallay
Framatome ANP
1911 North Ft. Myer Drive, Suite 705
Rosslyn, Virginia 22209

Mr. William A. Passetti, Chief
Department of Health
Bureau of Radiation Control
2020 Capital Circle, SE, Bin #C21
Tallahassee, Florida 32399-1741

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

Mr. Craig Fugate, Director
Division of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

Chairman
Board of County Commissioners
Citrus County
110 North Apopka Avenue
Inverness, Florida 34450-4245

Mr. Michael J. Annacone
Engineering Manager
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Mr. Daniel L. Roderick
Director Site Operations
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Senior Resident Inspector
Crystal River Unit 3
U.S. Nuclear Regulatory Commission
6745 N. Tallahassee Road
Crystal River, Florida 34428

Mr. Terry D. Hobbs
Manager, Nuclear Assessment
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, Florida 34428-6708

David T. Conley
Associate General Counsel II - Legal Dept.
Progress Energy Service Company, LLC
Post Office Box 1551
Raleigh, North Carolina 27602-1551