October 31, 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
            PROBABILISTIC METHODOLOGY FOR TUBE END CRACK ALTERNATE
            REPAIR CRITERIA (TAC NO. MC5813)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 222 to Facility Operating License
No. DPR-72 for Crystal River Unit 3. The amendment consists of changes to the existing
Technical Specifications in response to your letter dated January 27, 2005, as supplemented by
letters dated August 12, September 9, and October 21, 2005.

The amendment allows the licensee to utilize a probabilistic methodology to determine the
contribution to main steamline break leakage rates for the once-through steam generator
(OTSG) from the tube end crack (TEC) alternate repair criteria described in Improved Technical
Specification (ITS) 5.6.2.10.2.f and also involves a change to ITS 5.6.2.10.2.f to incorporate the
basis of the proposed probabilistic methodology and the method and technical justification for
projecting the TEC leakage that may develop during the next operating cycle following the
inservice inspection of each OTSG.

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. 222 to DPR-72
2. Safety Evaluation

cc w/encls: See next page

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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. 222
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 January 27, 2005, as supplemented by letters dated August 12, September 9, and October 21, 2005, 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. 222, 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 60 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: October 31, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 222

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

Remove	Insert
5.0-14A	5.0-14A
5.0-15	5.0-15
5.0-27	5.0-27
5.0-28	5.0-28
5.0-29	5.0-29

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 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 to the U.S. Nuclear Regulatory Commission (NRC) dated January 25, 2005, as supplemented by letters dated August 12, September 9, and October 21, 2005, Florida Power Corporation submitted a license amendment request for Crystal River Unit 3 (CR-3) regarding the implementation of the alternate tube criteria (ARC) for the tube end crack (TEC) degradation mechanism affecting the steam generator (SG) tubes. This amendment request involves changes to the method for evaluating the primary to secondary leakage which potentially may occur during a main steam line break (MSLB). Specifically, the amendment request would add the following text to the CR-3 Improved Technical Specifications (ITS) 5.6.2.10.2.f:

> The contribution to MSLB leakage rates from TEC indications shall be determined utilizing the methodology in Addendum B dated August 10, 2005 to Topical Report BAW-2346P, Revision 0. The projection of TEC leakage that may develop during the next operating cycle shall be determined using the methodology in Addendum C dated August 30, 2005 to Topical Report BAW-2346P, Revision 0.

The supplement dated August 12, 2005, revised the original proposed initial no significant hazards consideration determination that was published on March 15, 2005 (70 FR 12746) and the NRC published a revised notice in the *Federal Register*. The supplemental letters dated September 9, and October 21, 2005, provided clarifying information that did not expand the scope of the original application or change the proposed no significant hazards consideration determination published August 26, 2005.

2.0 BACKGROUND

CR-3 is a two loop pressurized-water reactor (PWR) with once-through SGs (OTSGs) manufactured by Babcock & Wilcox. OTSGs differ from recirculating SGs of the type manufactured by Westinghouse and other manufacturers, in that the tubes are straight (no u-bends) extending from the tube inlets at the upper tubesheet to the tube outlets at the lower tubesheet. Both ends of each tube are roll expanded against the respective tubesheet over a 1-inch length and are seal welded to the respective tubesheet at the tube end. The tubes are Mill Annealed Alloy 600.

On October 1, 1999, the NRC approved changes to the CR-3 technical specifications (TS) for an ARC applicable to SG tubes with axially oriented TEC indications within the upper and lower tubesheet areas. These changes provided, in part, "Tubes with axially oriented TEC may be left in service using the method described in Topical Report BAW-2346P, Revision 0, provided the combined projected leakage from all primary to secondary leakage, including axial TEC indications left in service, does not exceed the Main Steam Line Break (MSLB) accident leakage limit of one gallon per minute, minus 150 gallons per day, per OTSG." Under these approved changes, Topical Report BAW-2346P, Revision 0, includes an Addendum A, "CR-3 Plant Specific Leak Rates," dated May 28, 1999. This addendum modifies the generically applicable approach in the topical report for calculating leakage during MSLB to reflect CR-3 specific loads.

The condition monitoring assessment conducted as part of the SG inspections during Refueling Outage 13 (October 2003) identified that the calculated leak rate for a postulated MSLB based on the as found indications exceeded not only what had been projected during the operational assessment performed subsequent to Refueling Outage 12, but also exceeded the MSLB leakage limit.

The licensee is now proposing to revise its leakage assessment methodology to reflect (1) a revised, more realistic (less conservative) model for calculating leakage for a given set of flaws under a postulated MSLB and (2) a revised, more conservative methodology for projecting leakage under postulated MSLB occurring at the end of the next inspection interval. Taken together, these changes are intended to ensure that MSLB leakage assessments are conservative, yet also ensure that the potential for leakage under postulated MSLB are maintained to within acceptable limits.

3.0 REGULATORY EVALUATION

SG 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 SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 state 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, 10 CFR 50.55a specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code). Section 50.55a further requires, in part, that throughout the service life of a PWR [pressurized-water reactor] 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 [ISI] 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 ISI of SG tubing are augmented by additional SG tube surveillance requirements in the TS.
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 MSLB. These analyses consider the primary-to-secondary leakage through the tubing, which is assumed to 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, GDC-19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits).

Under the plant TS SG surveillance program requirements, the licensee is required to monitor the condition of the SG tubing and to plug or repair tubes as necessary. Specifically, the licensee is required to perform periodic inspections of and to repair or remove from service by plugging all tubes found to contain flaws with sizes exceeding the acceptance limit, termed "plugging limit" or "repair criteria." The specified plugging limits include a generally applicable depth-based limit equal to 40 percent of the initial tube wall thickness. For TEC degradation, an alternative repair criteria (plugging limit) may be applied as described above. The tube repair criteria (plugging limits) were developed with the intent of ensuring that degraded tubes (1) maintain factors of safety against gross rupture consistent with the plant design basis (i.e., consistent with the stress limits of the ASME Code, Section III) and (2) maintain leakage integrity consistent with the plant licensing basis while, at the same time, allowing for potential flaw size measurement error and flaw growth between SG inspections. The required frequency and scope of tubing examinations and the tube plugging limits are specified in ITS 5.6.2.10, "Steam Generator (SG) Tube Surveillance Requirements."

The subject TS amendment request concerns the accident leakage assessment methodology, which is implemented as part of the TEC ARC. The regulatory standard by which the NRC staff has evaluated this proposed methodology is that it must provide reasonable assurance that primary to secondary leakage that may occur as a consequence of a postulated accident is within the values assumed in the licensing bases accident analyses referred to above.

4.0 TECHNICAL EVALUATION

4.1 Accident Leakage Prediction Model

4.1.1 Current Model

The current leakage prediction model, as defined in BAW-2346P, is a deterministic model relating leak rate as a function of tube to tubesheet joint tightness. Joint tightness is a function of several contributors. These contributors include the interference fit associated with the radial roll expansion of the tubing against the tubesheet during fabrication, differential thermal expansion between the tubes and tubesheet under hot conditions (i.e., the tubes expand relative to the tubesheet increasing tightness), radial expansion of the tube relative to the tubesheet due to pressure inside the tube, distortion of the tubesheet holes (which may act to tighten or loosen the joints depending on circumstances) caused by tubesheet bow under primary to secondary pressure loads, and radial contraction of the tubing due to Poisson's effect under axial tensile load. The model uses "tube to tubesheet hole delta dilation (delta dilation)" as a surrogate parameter for joint tightness. Delta dilation is the relative difference in tubesheet bore and tube diameter dilation (radial expansion).

As documented in BAW-2346P, a test matrix was conducted to establish leak rate as a function of tube axial crack length, internal pressure, axial load, and delta dilation. Internal pressures and axial loads ranged to values intended to be generically bounding for MSLB. For each delta dilation value considered in the test matrix, five joint specimens were leak tested. The leak rate for each delta dilation value was assumed to be the mean value of these five tests evaluated at the upper 95 percent confidence value. Leak rate was determined to be insensitive to crack length.

Finite element analyses were conducted to establish the general structural behavior of the OTSGs (including the axial loads in the tubes and delta dilations) under normal operating and accident conditions. The resulting axial loads and delta dilations varied as a function of each tube's radial location from the centerline of the tube bundle. For CR-3, a CR-3 specific analysis was performed, as documented in Addendum A to BAW-2346P, with axial loads varying to a maximum of 660 lbf in the faulted OTSG and 1080 lbf in the intact OTSG under MSLB conditions. Because the test program showed leak rate to be a direct function of delta dilation (for a given pressure and maximum axial load) and because delta dilation was shown by the finite element analysis to be a function of tube radial location, the licensee was able to relate a leakage rate at CR-3 under MSLB conditions to tube radial location as documented in Addendum A to BAW-2346P. To simplify the utility of this relationship, the leakage rates are presented as a function of radial intervals where the corresponding leak rates are the maximum values within each interval. Thus, for a given set of TEC indications, each TEC indication is assigned an MSLB leak rate based on its radial location. The total MSLB leak rate is simply the sum of the leak rates for each individual indication.

4.1.2 Revised Model

The proposed revisions to the leakage prediction model include:

1. Adjustment to the leak rate versus delta dilation relationship stemming from the above mentioned test program.
2. Adjustment to the delta dilation versus tubesheet radius relationship for CR-3.
3. Use of a probabilistic model relating leak rate to delta dilation.

4.1.2.1 Adjustment to Leak Rate versus Delta Dilation

Even though a bounding axial load (3060 lbf) was applied during the above mentioned leakage tests, the licensee states that calculated delta dilations for the leak tests did not actually reflect the effect of axial load on delta dilation (due to axial load's Poisson's ratio effect on tube diameter); therefore, the tested joints were actually looser (i.e., greater tube-to-tubesheet delta dilation) than indicated by the calculated test delta dilation values. This yielded a conservative (high) leak rate for a given delta dilation. The licensee has now adjusted the delta dilation for each leakage test to include the incremental delta dilation produced by the axial load applied during the tests. Even with this adjustment, the test data show that leak rates under 3060 lbf axial load as a function of the adjusted delta dilations are conservative relative to leak rates under 1660 lbf axial load as a function of unadjusted delta dilations. The load value of 1660 lbf bounds the maximum CR-3 MSLB loads of 663 lbf and 1080 lbf for the faulted and unfaulted OTSGs, respectively. Thus, the NRC staff concludes that the proposed adjustment to MSLB leak rate versus delta dilation is conservative for CR-3.

4.1.2.2 Adjustment to Delta Dilation versus Tubesheet Radius Relationship

The current CR-3 specific delta dilation versus tubesheet radius relationship, as determined by finite element analyses, is given in Addendum A to BAW-2346P. These delta dilations considered the effects of CR-3 specific tubesheet distortion under primary to secondary pressure, tube to tubesheet differential thermal expansion, and free (non-end capped) pressure inside the tube, but did not consider the effect of axial load on tube dilation, which would tend to increase delta dilation slightly. The licensee stated that this was appropriate since the effects of axial load on tube dilation were also not considered during the tests performed to establish the leak rate versus delta dilation relationship, as discussed above. The licensee has now adjusted the delta dilation versus tubesheet radius relationship to include the incremental delta dilation produced by the CR-3 specific axial load. The equation used to make this adjustment is the same as that used to adjust the leakage test dilations. Thus, the NRC staff concludes that the proposed adjustment to delta dilation versus tubesheet radius relationship is conservative for CR-3.

4.1.2.3 Probabilistic Model Relating Leak Rate to Delta Dilation

The current deterministic leak rate model bins the leakage test data for each delta dilation interval (and tubesheet radius interval). Within each delta dilation (or radius) interval, leak rate for a given indication is assumed to be the mean value of the data within that interval, evaluated at an upper 95-percent confidence value. Total leak rate for a population of TEC indications is simply the summation of the calculated leak rates for each indication.

The NRC staff's evaluation of the proposed probabilistic model (termed leak TEC) was performed with the assistance of an expert statistician from Pacific Northwest National Laboratory (under a technical assistance contract to the NRC). The NRC staff finds that the proposed probabilistic model is very similar to probabilistic burst versus voltage and leak rate versus voltage models accepted by the NRC staff in Generic Letter 95-05, "Voltage Based Repair Criteria for Westinghouse Steam Generator Tube Affected By Outside Diameter Stress Corrosion Cracking." The model employs a log-normal regression fit to all the individual leakage data points as a function of delta dilation. The model includes consideration of the standard error of regression where the log of leak rate data about the regression line is assumed to be normally distributed. The model also considers the uncertainties of the regression line slope and intercept (expressed in terms of a variance-covariance matrix). The NRC staff and its consultant find that the log-normal regression model fits the data well and that the leakage data does appear to be log normally distributed.

The leak TEC model employs Monte Carlo simulations to generate leakage estimates for each crack and to determine total leakage at the desired probability and confidence level. Values of regression line slope and intercept and regression error are sampled during each Monte Carlo trial. For each indication, these values are used along with a random normal deviate applied to the regression error to generate a leak rate estimate. These estimates are summed to generate an estimate of total leak rate. This process is repeated 10,000 times, yielding a distribution of total leak rate estimates. The acceptance limit on MSLB leakage is applied to the one sided upper 95-percent probability/95-percent confidence value from that distribution. (The NRC staff's consultant recommends that this 95/95 estimate on total leak rate can better be termed as a "95-percent upper prediction interval or bound" on total leak rate.) The NRC staff notes that use of the 95/95 bounding estimate on total leak rate for determining acceptability is

equivalent to that found acceptable in NRC Generic Letter 95-05. The NRC staff finds the leak
TEC methodology, including the use of the 95/95 bounding estimate on the total SG leak rate,
to be acceptable.

4.2 Methodology for Projecting the Number of TEC Indications

An important aspect in ensuring that the amount of primary-to-secondary leakage during
postulated accident conditions is within acceptable levels is to be able to conservatively project
the number of indications that may exist at the end of the operating cycle, immediately prior to
the next scheduled SG inspection. This projected population of indications is assumed to leak
at the rates determined by the leakage prediction model evaluated in Section 4.1.2.3 of this
Safety Evaluation. For the TEC alternate repair criteria, past inspection data indicated that a
probability of detection (POD) adjustment of 0.84 was not adequate to conservatively project
the number of TEC indications at CR-3. As a result, a new methodology for accounting for the
number of TEC indications was developed by the licensee. This methodology is documented in
Addendum C to BAW-2346P, Revision 0.

Although the licensee's proposed methodology focuses on the leakage from several sources,
this leakage is a direct function of the number of indications and the radial distribution of the
indications within the SG. The proposed methodology for determining the total amount of TEC
leakage at the end of the next operating cycle is based on combining the leakage from the
following sources: (1) leakage from detected flaws left in service (as-left TEC leakage),
(2) leakage from flaws that were not detected (POD leakage), (3) a leakage increase as a result
of the difference between the as-left leakage from the prior outage and the as-found leakage in
the current outage (new leakage), and (4) any leakage increase as a result of an increase in the
rate of new leakage (additional leakage).

To evaluate the licensee's proposal, the NRC staff developed alternative methodologies for
projecting the amount of accident induced leakage. Some of these alternative methods
resulted in projected leakage values, which were more conservative than the results from the
licensee's method whereas others were less conservative. In general, the licensee's method
provided comparable results compared to the NRC staff's models. In addition to the above, the
NRC staff confirmed that the licensee's proposed methodology would have conservatively
projected the leakage observed during the licensee's 2003 outage had it been implemented
during their 2001 outage.

The licensee's methodology relies on several assumptions. These assumptions include that
the trend in new leakage will be similar from one cycle to the next (i.e., that the distribution of
new indications as a function of tubesheet radius is identical between successive operating
intervals) and that the rate of increase in leakage is linear. Given that these assumptions are
important in ensuring that the accident induced leakage is conservatively projected, the
licensee has proposed to report to the NRC the results of its assessment of the adequacy of the
predictive methodology following each SG tube inspection. If the assessment indicates that the
assumptions can not be fully supported, proposed corrective actions would also be reported to
the NRC.

Based on the above, the NRC staff concludes that the licensee's proposed changes to the
leakage methodology are acceptable. The NRC staff also concludes that the proposed

changes to the reporting requirements are acceptable. These latter changes will permit the NRC staff and the licensee to confirm the continued adequacy of the methodology.

4.3 NRC Staff Conclusion

The NRC staff finds that the proposed TS amendment ensures that the leakage integrity of the tube-to-tubesheet joints with TEC will be maintained consistent with the assumptions employed in the licensing basis accident analyses and, thus, in accordance with the applicable regulations without undue risk to public health and safety.

5.0   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.

6.0   ENVIRONMENTAL CONSIDERATIONS

The amendment changes surveillance requirements. 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 (August 26, 2005 (70 FR 50424)). 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.

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

Principal Contributors: E. Murphy, NRR
                                    K. Karwoski, NRR

Date: October 31, 2005

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Florida Power Corporation

cc:

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