August 30, 2001

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 CONTAINMENT LEAKAGE RATE TESTING PROGRAM (TAC NO. MB1349)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 197 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 March 7, 2001, as supplemented April 25, 2001, June 20, 2001, and July 16, 2001. The amendment provides for an alternate method for complying with the requirements of Title 10, *Code of Federal Regulations* (10 CFR) Section 50.54(o) and 10 CFR Part 50, Appendix J, Option B. Specifically, the amendment allows a one-time interval increase for the CR-3 Type A, Integrated Leakage Rate Test for no more than 5 years.

The staff requested additional information by letter dated July 6, 2001, and Florida Power Corporation responded to this request by letter dated July 16, 2001.

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/

John M. Goshen, 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. 197 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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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. 197
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 March 7, 2001 as supplemented April 25, June 20, and July 16, 2001, 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. 197, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the
Technical Specifications

Date of Issuance: August 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following page of the Appendix "A" Technical Specifications with the attached page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

Remove	Insert
5.0-23A	5.0-23A

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated March 7, 2001, as supplemented April 25, 2001, June 20, and July 16, 2001, Florida Power Corporation (FPC), the licensee for the Crystal River Unit 3 (CR-3) requested a Technical Specification change that would allow a one-time change in their Appendix J, Type A, test interval from the required 10 years to 15 years. Without an extension, the licensee would have to perform a Type A test during their next refueling outage, scheduled to begin in fall 2001. The NRC requested additional information by letter dated July 6, 2001, and FPC responded to this request by letter dated July 16, 2001. The April 25, June 20, and July 16, 2001, letters provided clarifying information and did not expand the scope of the original *Federal Register* notice.

2.0 BACKGROUND

Title 10, *Code of Federal Regulations* (10 CFR), Part 50, Appendix J, Option B, requires that a Type A containment integrated leakage rate test (ILRT) be conducted at a periodic interval based on historical performance of the overall containment system. CR-3 Technical Specification (TS) 5.6.2.20 requires that a program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. It further requires that this program shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995. This RG references Nuclear Energy Institute (NEI) 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months under certain circumstances.

The most recent two Type A tests at CR-3 have been successful, so the current interval requirement is 10 years.

FPC requested an addition to TS 5.6.2.20, “Containment Leakage Rate Testing Program,” which would indicate that they are allowed to take an exception to the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS says that the first Type A test performed after the November 7, 1991, Type A test shall be performed no later than November 6, 2006. This would make the interval 15 years since the last test.

FPC states that the requested extension would allow them to move the Type A test to a subsequent refueling outage, when it could be performed off the critical path of outage activities, thereby saving $1.4 million in replacement power and daily outage support costs.

3.0 EVALUATION

3.1 CONTAINMENT SYSTEMS EVALUATION

FPC has performed a risk impact assessment of extending the Type A test interval to 15 years. The assessment was provided to the U.S. Nuclear Regulatory Commission (NRC) in a June, 20, 2001, letter from FPC (ADAMS Accession No. ML011780339). In performing the risk assessment, they considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) TR-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing,” and RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.”

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, “Performance-Based Containment Leak-Test Program,” September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC’s rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from three per 10 years to one per 10 years, will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of leakage, in percent of person-rem/year, for the pressurized water reactor representative power plant was estimated to increase from .032 percent to .035 percent. This confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three per 10 years to one per 10 years leads to an imperceptible increase in risk.
Building upon the methodology of the EPRI study, FPC assessed the change in the predicted person-rem/year frequency. The NRC considers FPC’s assessment an improvement of the EPRI study because the leakage from sequences that have the potential to result in large releases if a pre-existing leak were present were quantified. Since the Option B rulemaking in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment in risk-informed changes to a plant’s licensing basis. The licensee has proposed using RG 1.174 to assess the

acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF the relevant criterion is the change in LERF which the licensee estimated. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. FPC estimated the change in the conditional containment failure probability to show that the defense-in-depth philosophy is met.

FPC examined plant specific accident sequences from their Individual Plant Examination. The following sequences were considered in the assessment:

- Core damage sequences in which the containment remains intact initially and in the long term.

- Core damage sequences in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress. These sequences involve Type A tests and potential failures not detectable by local leak rate tests (e.g., a hole in the containment liner). The impact on risk from changes in Type A test frequency are evaluated by investigating these sequences.

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left 'opened' following a plant post-maintenance test. For example, valve failing to close following a valve stroke test.

- Accident sequences in which the containment is bypassed or involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents. Therefore, these sequences were not evaluated further after their initial quantification.

The steps taken by the licensee to perform the risk assessment are as follows:

- Quantified the base-lined risk in terms of frequency per reactor year for each of the eight accident classes evaluated in EPRI TR-104285.

- Developed plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

- Evaluated the risk impact of extending the Type A test interval from 10 to 15 years and the cumulative impact of extending the interval from the original three per 10 years to 15 years.

- Determined the change in risk in terms of LERF in accordance with RG 1.174.

Determining the change in risk in terms of LERF involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in a large release due to failure to detect a pre-existing leak during the extension period. FPC designated these sequences as Class 3B sequences and estimated a frequency of 2.90 x 10^{-7}/year, based on the original 3-year test interval. FPC then used the EPRI methodology to estimate the impact of the Type A test interval on the leakage probability. Extending the Type A test interval from the original test interval to 10 years increases the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. For a 10-year interval there is a 10 percent increase in the overall probability of leakage (3 * 60/18) versus 15 percent for a 15-year interval. FPC estimated a Class 3B sequence frequency of 3.19 x 10^{-7}/year for the 10-year interval and 3.34 x 10^{-7}/year for the 15-year interval. Therefore, the increase in LERF can be estimated by the change in the frequency of Class 3B sequences. Extending the Type A test interval from the current 10-year interval to 15 years resulted in a 1.5 x 10^{-8}/year increase. If the risk increase is measured from the original three per 10 year interval, the increase in LERF is 4.4 x 10^{-8}/year.

The following conclusions can be drawn from FPC's risk assessment associated with extending the Type A test frequency:

1. The risk assessment predicted a slight increase in risk when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk (person-rem/year within 50 miles) was found to be 0.045 percent. The increase in the total integrated plant risk, given the change from a three per 10-year test interval to a 15-year test interval, was found to be 0.14 percent. This is reasonable when compared to the range of risk increase, 0.02 to 0.14 percent, estimated in NUREG-1493 when going from a three per 10-year test interval to a 10-year interval. NUREG-1493 concluded that a reduction in the frequency of tests from three per 10 years to one per 10 years leads to an imperceptible increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in CDF less than 10^{-6} per reactor year and increases in LERF less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from one per 10 years to one per 15 years is estimated to be 1.5 x 10^{-8}/year. The increase in LERF resulting from a change in the Type A test interval from the original three per 10 years to one per 15 years is estimated to be 4.4 x 10^{-8}/year. Increasing the Type A interval to 15 years is considered to be a very small change in LERF.
3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in the conditional containment failure probability was estimated to be an increase of 0.0015 for the proposed change and 0.0031 for the cumulative change of going from a test interval of three per 10 years to one per 15 years. The NRC finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed change.

The NRC recognizes the limitations of a conditional containment failure probability approach. For plants, such as Crystal River, with core damage frequency estimates well below 10^{-4}, the ability of the containment to withstand events of even lower probability becomes less clear. Therefore, it is important to consider other risk metrics in conjunction with the conditional containment failure probability, such as total LERF. FPC's submittal has sufficiently demonstrated that the total LERF is less than 10^{-5}, and the NRC finds this acceptable.

3.1.1 CONTAINMENT SYSTEMS EVALUATION CONCLUSION

Based on these conclusions, the NRC finds that the increase in predicted risk due to the proposed change is within the acceptance criteria while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

3.2 CONTAINMENT INTEGRITY EVALUATION

The CR-3 containment pressure boundary consists of the containment structure, containment access penetrations, and other process piping and electrical penetrations. The integrity of the penetrations is verified through Type B and Type C local leak rate tests (LLRTs) as required by 10 CFR Part 50, Appendix J, and the overall integrity of the containment structure is verified through an ILRT. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design basis accident pressure. The last ILRT for CR-3 was performed in November 1991. With the extension of the ILRT time interval, the next overall verification will be performed in 2006. Because the ILRT, the LLRTs, and inservice inspection (ISI) of the containment in combination ensure the leak-tight and structural integrity of the containment, the staff requested additional information regarding the ISI of containment and potential areas of weaknesses in the containment that may not be apparent in the risk assessment. The following is an evaluation of FPC's responses to the NRC's request for additional information (RAI) dated
July 6, 2001.

FPC is using the 1992 Edition and the 1992 Addenda of Subsections IWE and IWL of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code) for conducting its inservice inspection of the CR-3 containment with certain approved relief from some Code requirements. The ISI interval began on August 14, 1998, and will end on August 7, 2008. FPC's responses indicate that the accessible areas of the containment pressure boundary will be periodically monitored for signs of degradations.

Question 5 of the RAI was related to the effects of degradations in uninspectable areas of the steel liner (i.e., areas that cannot be visually examined). However, ILRTs may identify major areas of through-wall degradations when the containment is pressurized. In response to Question 5, FPC explained that the potential for containment leakage is explicitly included in the risk assessment. By definition, the intact containment cases (EPRI Containment Failure Class 1) include a leakage term, which is independent of the source of the leak. The CR-3 Individual Plant Examination (IPE) was submitted to the NRC on March 9, 1993 (3F0393-03) and approved by the NRC on June 30, 1998 (Accession No. M9807200250). Section 4.4 of the IPE examined and evaluated the containment failure mode analysis results and developed integrated containment failure probability distributions. The analysis identified seven failure locations that would result in a large leakage area and quantified the expected failure pressure of each location for use in the IPE. This analysis was then utilized in the development of the IPE source terms. The IPE source terms were utilized as input to the Generic Level 3 Probabilistic Risk Assessment for CR-3 (BAW-2369) and was submitted to the NRC on April 25, 2001, as Attachment C. The doses from this assessment were utilized in the FPC Calculation F-01-0001, Revision 2, Evaluation of Risk Significance of ILRT Extension, and was submitted to the NRC on June 20, 2001.

The NRC's review of Section 4.4 of the IPE found that FPC calculated the mean containment capacity for seven failure modes. The cumulative containment failure distribution at temperatures ranging from 300°F and 800°F indicate that at the conditional containment failure probability of 10% (signifying a large leakage), the non-degraded containment (i.e., no corrosion on the liner) can withstand an internal pressure of 100 psig. It should be recognized that the ASME Code allows liner corrosion up to 10% of the liner thickness with a requirement for monitoring the degradation during subsequent inspections. The ILRT pressure for CR-3 is 54.2 psig, about half the pressure that might result in a large leakage. If sufficient time is allowed for corrosion on the uninspectable side of the liner plate to continue, the ILRT is likely to result in appreciable leakage.

Overall, the NRC finds that (1) the containment structural integrity is verified through periodic ISIs conducted as required by Subsections IWE and IWL of the ASME Code, Section XI, (2) the potential for large leakage from the areas that cannot be examined by the ISI has been explicitly modeled in performing the risk assessment, and (3) the integrity of the penetrations and containment isolation valves are periodically verified through Type B and Type C tests as required by the CR-3 TS. Moreover, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair replacement activities in accordance with Article IWE-5000 of the ASME Code, Section XI. Serious degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 and 10 CFR 50.73.

3.2.1 CONTAINMENT INTEGRITY EVALUATION CONCLUSION

On the basis of the above findings, the NRC finds that a one-time extension of performing the ILRT as proposed by the licensee in Section 5.6.2.20 of the proposed TS amendment request 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. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATIONS

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 and 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 (66 FR 17967). 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

Based on the foregoing evaluation, the staff finds that the interval until the next Type A test at CR-3 may be extended to 15 years, and that the proposed changes to TS 5.6.2.20 are acceptable.

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: Michael Snodderly, James Pulsipher, Hansrai Ashar, NRR

Date: August 30, 2001

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