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

April 13, 1999

SUBJECT: CRYSTAL RIVER UNIT 3 - STAFF EVALUATION AND ISSUANCE OF AMENDMENT RE: MAKEUP SYSTEM LETDOWN LINE FAILURE ACCIDENT ANALYSIS (TAC NO. M99571)

Dear Mr. Cowan:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 173 to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). This amendment is in response to a Florida Power Corporation request dated September 9, 1997, and supplemented by letters dated November 7 and November 25, 1997, and January 20 and October 30, 1998, in which you proposed to revise the Final Safety Analysis Report (FSAR) analysis of the Makeup System letdown line failure accident. The revised analysis models the event as being terminated by manual operator action to isolate the line whereas the original analysis models an automatic isolation of the break.

The amendment approves changes to the FSAR, and requires that the changes be submitted with the next update of the FSAR pursuant to 10 CFR 50.71(e). A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

L. Wiens, 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.173 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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

April 13, 1999

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UNITED STATES
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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.
CITY OF TALLAHASSEE
DOCKET NO. 50-302
CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
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 September 9, 1997, as supplemented November 7 and November 25, 1997, and January 20 and October 30, 1998, 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, changes to the updated Final Safety Analysis Report (FSAR) to reflect changes to the Makeup System Letdown Line Failure Accident analysis at Crystal River Unit 3, as set forth in the application for amendment by Florida Power Corporation dated September 9, 1997, and supplemented November 7 and November 25, 1997, and January 20 and October 30, 1998, are authorized. The licensee shall submit the revised description authorized by this amendment with the next update of the FSAR in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented as specified in (2) above.

FOR THE NUCLEAR REGULATORY COMMISSION

Sheri R. Peterson

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

Date of Issuance: April 13, 1999

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-72 INVOLVING A CHANGE TO THE MAKEUP SYSTEM LETDOWN LINE FAILURE ACCIDENT ANALYSIS

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

1.0 Introduction

By letter dated September 9, 1997, Florida Power Corporation (FPC/or the licensee) submitted a request for an amendment to its facility operating license for Crystal River Unit 3 (CR-3). Additional information was submitted by letters dated November 7 and November 25, 1997 and January 20 and October 30, 1998 which did not affect the original proposed no significant hazards consideration determination, or expand the scope of the request as noticed in the Federal Register. The amendment request addresses an analysis revision for the makeup system letdown line failure accident as discussed in the Final Safety Analysis Report (FSAR). In the original analysis, this event was modeled as being terminated by an automatic isolation of the failed letdown line on low reactor coolant system pressure. During Refueling Outage 10, the licensee determined that symptom-oriented actions in the Emergency Operating Procedures (EOP) could prevent the automatic isolation of the letdown line as assumed in the design basis analysis. The revised analysis has modeled the event as being terminated by manual operator action to isolate the line. Since the delay in isolation would increase the amount of radioactivity released, radiation doses to persons offsite and in the control room could increase. On this basis, FPC determined that the change involves an unreviewed safety question.

2.0 Background

The licensee stated in its September 9, 1997, submittal that before Revision 23 of its FSAR, the Makeup System Letdown Line Failure Accident analysis assumed that loss of reactor coolant outside containment would be terminated by an automatic Engineered Safeguards Actuation System (ESAS) isolation signal when the low reactor coolant system (RCS) pressure set point was reached. The signal would close the containment isolation valves for the letdown line and the leak would be terminated.

During Refueling Outage 10, CR-3 discovered that guidance in EOP-3, "Inadequate Subcooling Margin," prevented automatic isolation of the letdown line from occurring under a certain sized line failure because the EOP directs the operator to ensure full high pressure injection (HPI) system flow upon a loss of subcooling margin. HPI would repressurize the RCS thereby preventing a decrease in pressure to the low RCS pressure set point, which would prevent automatic isolation of the letdown line from occurring because no ESAS signal would be received. The licensee indicated that the sequence of steps in the EOP was based upon the guidance provided by the Babcock and Wilcox (B&W) Owners Group Emergency Procedures Technical Bases Document (Abnormal Transient Operating Guidelines [ATOG]), a document approved by the NRC.

As a result, the licensee has reanalyzed the event, now crediting the operators for isolation of the letdown line. The analysis now assumes that the break occurs in the letdown line which results in the RCS pressure dropping. The makeup system is assumed to respond to the loss of inventory and increase to full flow. This slows the RCS depressurization and maximizes the calculated inventory loss out the failed letdown line. The RCS pressure decreases to the point where a saturation condition exists in the hot leg, the loss of subcooling margin setpoint is reached and the operators are alerted to the condition. The analysis assumes that it then takes the operators 10 minutes to manually initiate HPI and isolate the letdown with redundant isolation valves. The analysis assumes that inventory is lost the entire time from the initiation of the break to ten minutes following the loss of subcooling in the hot leg.

3.0 EVALUATION

3.1 Analysis Methodology

To evaluate the radiological consequences, an analysis was performed to determine the total mass of primary inventory that is released into the auxiliary building following the failure of the letdown line. Although the original analysis was performed with CRAFT2, the new analysis was performed with RELAP5/MOD2-B&W which is described in Topical Report BAW-10192P, "BWNT Loss-of-Coolant-Accident Evaluation Model for Once-Through Steam Generator Plants." This topical report was reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) for loss-of-coolant accident (LOCA) analysis and is appropriate for calculating the loss of inventory associated with a letdown line failure (BAW-10192P-A submitted to the NRC by letter dated August 14, 1998). By letter dated October 30, 1998, the licensee addressed the NRC limitations and restrictions placed upon the use of the code. The submittal describes instances where changes were made to some of the modeling assumptions described in the topical report and in the NRC review of the methodology; however, the licensee described how the modeling assumptions chosen continued to yield conservative and acceptable results. As a result, use of the new methodology is acceptable for use in evaluating the mass deposited into the auxiliary building from a letdown line break.

The results of the new analysis predict 114,000 pounds-mass (lbm) of primary coolant deposited into the auxiliary building. This value is increased from 45,760 lbm in the old analysis. There are no acceptance criteria on total mass of primary inventory released into the auxiliary building; however, the radiological consequences must be acceptable. The staff has reviewed the licensee's submittal and determined that the calculated mass release, 114,000 lbm into the auxiliary building, is an appropriate value to use in evaluating the radiological consequences of this pipe break. The staff makes this determination based on the licensee's use of conservative inputs and assumptions in the analysis and the use of an NRC-approved methodology that is appropriate for this application.

3.2 Manual Actions

The original design of the licensee's safety systems and the systems' ability to respond to design-basis accidents are described in the licensee's FSAR and were reviewed and approved by the NRC. Automatic action is frequently provided as a design feature specific to each safety system to ensure that the specific functions of the system will be accomplished. In a few cases, limited operator actions, when appropriately justified, were approved. Proposed changes that substitute manual operator actions for automatic system actuation or modify existing operator actions, including operator response times, previously reviewed and approved during the original licensing review must be evaluated under the criteria of 10 CFR 50.59. In those instances where licensees consider temporary or permanent changes to the facility which credit operator actions, the NRC has relied on the guidance provided in Generic Letter (GL) 91-18, November 7, 1991, "Resolution of Degraded and Nonconforming Conditions and on Operability," and ANSI/ANS 58.8, "Time Response Design Criteria for Safety Related Operator Actions," 1994, for evaluating such changes.

The staff used the guidance in GL 91-18 and ANSI-58.8 relevant to manual operator actions and response times to perform those actions to complete its evaluation of the licensee's submittals dated September 9, 1997, November 7, 1997, and January 20, 1998. GL 91-18 states that in a "... situation in which substitution of manual action for automatic action may be acceptable, the licensee's determination of operability must focus on the physical differences between automatic and manual action and the ability of the manual action to accomplish the specified function. The physical differences to be considered include, but are not limited to, the ability to recognize input signals for action, ready access to or recognition of setpoints, design nuances that may complicate subsequent manual operation such as auto-reset, repositioning on temperature or pressure, timing required for automatic action, etc., minimum manning requirements, and emergency operating procedures written for the automatic mode of operation. The licensee should have written procedures in place and training accomplished on those procedures before substitution of any manual action for the loss of an automatic action." ANSI-58.8 provides estimates of reasonable response times for operator actions, and allows licensees to use time intervals derived from independent sources, provided they are based on analyses with consideration given to human performance.

The staff evaluated the licensee's revised analysis of substituting manual operator action for automatic isolation of the failed letdown line with regard to the following considerations: specific operator actions and the times to perform those actions; environmental conditions expected; procedural guidance for the required actions; support personnel and/or equipment required to carry out the required actions; specific operator training necessary to carry out the required actions; information requirements including qualified instrumentation; recovery from credible errors; and risk significance of the proposed operator actions.

To resolve the condition created by EOP-3 described in the "Background" section of this evaluation, the licensee revised EOP-3 to require manual isolation of the letdown line earlier in the procedure. As the licensee indicated in a Technical Specification (TS) change request dated June 14, 1997 (which was reviewed and approved by the NRC staff in a Safety Evaluation dated January 24, 1998), EOP-3 requires manual initiation of HPI on a Loss of Subcooling Margin (LSCM) if HPI was not already automatically initiated. This step is followed immediately by manually actuating reactor building isolation and cooling (RBIC) which isolates the failed letdown line.

In response to staff questions dated October 10, 1997, the licensee described the proposed manual action of isolating the failed letdown line in its November 7, 1997 submittal to the staff. In that submittal the licensee explained how the EOPs are used to implement the required action stating,

> "Isolating letdown is very simple. There are two pushbuttons on the engineered safeguards (ES) section of the main control board (MCB) to actuate manual RBIC. Each pushbutton closes one train of containment isolation valves. Thus, depressing just one of the two pushbuttons will isolate the letdown flow path. Rule 1 [Loss of SCM, EOP-13], requires both pushbuttons to be depressed."

In its analysis, the licensee assumed that the break (failed letdown line) will be isolated within 10 minutes after the hot legs reach saturation conditions. In its November 7, 1997, submittal, the licensee stated that,

> "All six operating crews plus backups have been trained as of October 24, 1997. The operating crews have consistently performed the required actions on the plant-specific simulator within the 10 minutes. ... A simulator run was performed on October 17, 1997 for a decay heat drop line break scenario. The scenario included LSM [loss of subcooling margin] and the steps were timed. The RBIC actuation was performed on both "A" and "B" sides within 72 seconds of loss of subcooling margin. This is representative of crew response."

The licensee further explained in its November 7, 1997, submittal that "adequate time would exist for the identification of an operator error and correction of this error before any significant increase in the consequence of this event would occur." The licensee further stated that the operator error assumed in its analysis is a failure to isolate the letdown line within 10 minutes of reaching saturation; the operator never isolating the line was not considered a credible event because of training, multiple procedure steps requiring letdown line isolation, and the inclusion of RBIC actuation which closes the letdown line containment isolation valves.

In addition, numerous indications and alarms are available to the operator of a loss of reactor coolant in the event of a postulated letdown line break such as decreasing RCS pressure (resulting in a reactor trip); decreasing pressurizer level; decreasing makeup tank level; increased makeup flow; and a loss of adequate subcooling. Radiation monitors in the auxiliary building ventilation stack; auxiliary building sump alarms; and visual and audible indications of a letdown line break would also provide timely indication to the operator that RCS leakage was occurring due to a letdown line break. The licensee also stated that, since the letdown line is equipped with multiple isolation valves, it can be isolated in the event of a single failure.

Based on the preceding information, the staff has determined that the proposed substitution of manual isolation for a previously automatic isolation of a failed letdown line will not adversely affect operator performance and is consistent with staff guidance and; therefore, is acceptable.

3.3 Dose Assessment

The staff reviewed the FPC's proposed analysis of the potential radiological consequences of the letdown line failure assuming isolation by manual operator action. The effect of the delay in break isolation increases the mass of reactor coolant released through the break. FPC determined that the mass release would be 114,000 lbm, an increase from 45,760 lbm. FPC's analysis indicated that offsite radiation doses would be within the dose guidelines of 10 CFR Part 100. FPC did not address the impact of the increased mass release on the doses received by control room personnel during the event. Also, FPC performed the radiological analyses assuming an RCS activity based on 1% failed fuel. This assumed activity is less than the maximum activity allowed by TS. These two issues were identified to FPC in a letter dated December 9, 1997. FPC responded to these issues in a letter dated January 20, 1998.

Representatives from FPC and the NRC staff met at NRC offices in Rockville, Maryland, on January 8, 1998, to discuss a justification for continued operation for the control complex habitability envelope (CCHE). FPC confirmed these discussions in a letter dated January 14, 1998. In that letter, FPC committed to perform a detailed CCHE evaluation and to submit a revised Control Room Habitability Report within 6 months of the restart from the then-current outage. This report was submitted by letter dated July 30, 1998. Included in this evaluation was a discussion of the control room doses resulting from a letdown line break with RCS activity levels consistent with the TS. The staff has reviewed the letdown line break evaluation described in this report and concluded that the analysis is acceptable and that there is reasonable assurance that the doses to control room personnel due to a letdown line break would be within the criteria of 10 CFR Part 50, Appendix A, GDC-19 and NUREG-0800 Section 6.4.

For offsite doses, FPC assumed that the RCS activity would be that resulting from 1% failed fuel. This exceeds the TS equilibrium activity limiting condition for operation (LCO) of 1.0 μCi/g dose equivalent (DE) I-131 specified in CR-3 LCO 3.4.15.A, but is less than that specified in LCO 3.4.15.B (effectively 60 μCi/g DE I-131). The RCS activity associated with 1% failed fuel is approximately 7 μCi/g DE I-131. The intent of LCO 3.4.15.B is to allow for operation at RCS activity greater than 1.0 μCi/g DE I-131 for limited periods of time to allow for cleanup of iodine spiking events.

The staff addresses iodine spiking in design basis analyses by a model that considers two cases. In the first, it is assumed that a pre-accident transient has caused the DE I-131 concentration to increase to 60 μCi/g. This concentration is the maximum reactor coolant activity allowed by LCO 3.4.15.B before a reactor shutdown is mandated. The second case involves a coincident spike which is quantified in terms of a release rate from fuel. The rate is calculated as 500 times the release rate which gives a reactor coolant activity equivalent to the LCO 3.4.15.A. Since the pre-incident spike case is consistent with LCO 3.4.15.B, the staff asked FPC to consider this case in a letter dated December 9, 1997. In its January 20, 1998, letter, FPC stated that consideration of RCS activity at a level of 60 μCi/g DE I-131 was not required by the CR - 3 licensing basis. FPC based this conclusion on the following: (1) the licensing basis analysis for the letdown line break for offsite consequences for CR-3 is based on one percent defective fuel rods and did not assume iodine spiking; and (2) its position that safety analyses do not assume design basis accidents are initiated when in an action statement allowed by the TS.

The staff considered FPC's position regarding LCOs in the context of the CR-3 TS bases and the standard technical specifications (STS) for B&W plants. The staff also reviewed the STS for the other two pressurized water reactor (PWR) vendors. The staff practice is to limit the radiological consequences of an event with RCS activity at 1.0 μCi/g DE I-131 to a small fraction of the dose guidelines of 10 CFR Part 100; but to use the full dose guidelines for an event with the pre-incident iodine spike activity. The increased acceptance criteria for the iodine spike case reflects the lower probability of a letdown line break occurring concurrent with a pre-incident spike. This practice was documented in NUREG-0800, which post-dated the issuance of the CR-3 operating license. The staff practice is also reflected in the bases of the STS for the PWR vendors other than B&W. However, the B&W STS bases and the CR-3 bases indicate that the action statement to allow limited operation with RCS activity greater than 1.0 μCi/g DE I-131 provides reasonable time for temporary activity increases to be cleaned up with processing systems, and that the safety analysis for an event with off site releases assumes one percent defective fuel. The staff accepts FPC's conclusion that, for this specific analysis, the assumption of 60 μCi/g DE I-131 is not consistent with the licensing basis for the letdown line break based on the language in the bases. However, in general, the staff does require that safety analyses for design basis accidents assume the most limiting parameters allowed by TS including those allowed by Action Statements.

The NRC staff performed independent calculations of a letdown line break assuming an RCS activity of 60 μCi/g DE I-131. The staff analysis assumed that 114,000 lbm of RCS water containing activity at 60 μCi/g DE I-131 would be available for release. The staff assumed that 10% of the radioiodine in the released liquid would become airborne. The remaining 90% of radioiodine in the released reactor coolant remains in the liquid phase and is collected in sumps for processing as liquid waste. All noble gases in the released coolant are assumed to become airborne. The staff reviewed the letdown system configuration and determined that the temperature of the released liquid would be less than 212 °F under design conditions. Based on a constant enthalpy process, less than 10% of the liquid would flash to vapor. The licensee evaluated two cases with regard to release filtration. In one case, no filter credit was taken. In the second case, credit for 90% reduction in the iodine release activity was taken. Since the auxiliary building ventilation system is not powered from emergency power sources, the staff only considered the no filtration case. The results of this calculation showed that doses would be a small fraction of the dose guidelines of 10 CFR Part 100. The staff has determined that the letdown line break evaluation described in FPC's Control Room Habitability Report was appropriate and that there is reasonable assurance that the offsite doses due to this letdown line break would be a small fraction of the dose guidelines of 10 CFR Part 100. Therefore, since the doses to control room personnel and the offsite doses are acceptable, the requested change to the CR-3 FSAR is acceptable.

4.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. NRC, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (62 FR 50005). 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 its review, the staff determined that the proposed change to the Makeup System Letdown Line Failure Accident analysis for Crystal River Unit 3 used conservative inputs and assumptions and used an NRC approved methodology that was appropriate for determining the mass release into the auxiliary building. The use of manual action to isolate the failed letdown line would not adversely affect operator performance and is consistent with staff guidance. There is reasonable assurance that the doses to control room personnel due to a letdown line break would be within the criteria of 10 CFR Part 50, Appendix A, GDC-19 and NUREG-0800 Section 6.4., and would be a small fraction of the dose guidelines of 10 CFR Part 100 and therefore the dose consequences resulting from this postulated break are acceptable. Therefore, the staff concludes that the requested change to the CR-3 FSAR is acceptable.

Principal Contributor: S. F. LaVie, J. Bongarra, C. Jackson

Date: April 13, 1999

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Florida Power Corporation

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