May 23, 1986 DMB 016

Docket No. 50-302

Mr. Walter S. Wilgus
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Florida Power Corporation
ATTN: Manager, Nuclear Licensing
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Dear Mr. Wilgus:

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the license in response to your application dated April 24, 1986.

This amendment permits revision of the design of the reactor coolant pump supports in accordance with the recent modifications to General Design Criterion 4 and the supplementary information contained in the notice of such modification.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Harley Silver, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 89 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

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Crystal River Unit No. 3 Nuclear
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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
SEBRING UTILITIES COMMISSION
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. 89
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 April 24, 1986, 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 the addition of the following license condition:

   2.C.(10) The design of the reactor coolant pump supports need not include consideration of the effects of postulated ruptures of the primary reactor coolant loop piping and may be revised in accordance with Florida Power Corporation's amendment request of April 24, 1986.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Date of Issuance: May 23, 1986

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

INTRODUCTION

On April 11, 1986, a notice of modification of General Design Criterion 4 (GDC-4) of Appendix A, 10 CFR Part 50, was published in the Federal Register. The modified rule, effective May 12, 1986, allows exclusion from the design basis of the dynamic effects of postulated ruptures in primary coolant loop piping when analyses demonstrate the probability of rupturing such piping is extremely low under design basis conditions. Specifically, leak-before-break (LBB) technology could be employed to demonstrate that low probability. This consists of the use of advanced fracture mechanics analysis and techniques to demonstrate the capability to detect leakage well before any cracks in the pipe wall can become unstable and grow to failure. Supplementary information in the notice indicates that an amendment to the license would be necessary to implement such changes in existing plants, and that guidance in that supplementary information should be used to apply the modified rule.

In February 1985 (Ref. 1), Florida Power Corporation (FPC) had requested a partial exemption from those portions of GDC-4 which require protection of structures, systems, and components against certain dynamic (including mechanical and structural loading) effects associated with postulated Reactor Coolant System (RCS) main loop pipe breaks. This exemption pertains to all postulated breaks specified in the Crystal River Unit 3 (CR-3) reactor coolant main loop piping. The request does not affect the CR-3 design basis for environmental, containment, equipment qualification or Emergency Core Cooling System (ECCS) analysis. FPC has proposed to utilize advanced fracture mechanics techniques (LBB) at CR-3 to eliminate postulated RCS main loop pipe breaks and consequently their inherent mechanical and structural load effects.

In Ref. 3, Babcock & Wilcox (B&W) submitted for NRC staff review a fracture mechanics analysis to validate the LBB failure scenario for their Nuclear Steam Supply System designs. NRC staff review and approval of this analysis was documented in Ref. 4, and concluded that "the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of the B&WOG facilities is sufficiently low such that dynamic effects associated with postulated pipe breaks in these facilities (including CR-3) need not be a design basis."

In Refs. 5-7, FPC submitted detailed information and an evaluation of revised RC pump support configurations, which are based on the elimination of the RCS pipe loads due to postulated breaks. Based on this approach, FPC has indicated that the 32 large bore snubbers currently restraining the four RC pumps can be replaced by four smaller snubbers and four struts. In Ref. 10, FPC provided an assessment of the CR-3 leak detection system.

By letter dated April 24, 1986 (Ref. 10), FPC requested amendment to Facility Operating License No. DPR-72 for CR-3 based on the documentation previously submitted in support of its exemption request. The proposed amendment would add a license condition stating that the design and licensing basis of the reactor coolant pump supports need not include consideration of the effects of postulated ruptures of the primary reactor coolant loop piping.

EVALUATION

The Supplementary Information in the April 11, 1986 notice of modification of GDC-4 provides guidance that the following should be performed in applying the modified rule:

- Plant unique analyses to demonstrate adequate margins for all remaining loads.
- Confirmation that CR-3 falls within the vendor-calculated envelope.
- Demonstration of improved overall system performance and reliability compared with the previous support system.
- Consideration of independent design and fabrication verification.
- Demonstration that leak detection capability is adequate.

We reviewed the licensee's submittals with regard to the above guidance. Our evaluation is as follows.

A. Analysis

FPC has proposed to reduce the number of restraints on each RC pump from a total of eight large bore snubbers to two restraints, consisting of combinations of smaller snubbers and struts. FPC has submitted the results of a structural analysis of the RCS with the reduced number of restraints subjected to dead weight, thermal expansion, and earthquake loads. (This analysis was actually performed by B&W). The results showed that adequate margin exists with respect to allowable stresses and fatigue usage factors.

Based on the information provided in these submittals, a number of uncertainties and concerns were identified which required clarification and resolution before approval of the requested amendment. The major concerns which we identified were as follows:

1. The basis for the modeling of the RCS and the reactor building interior concrete, including the associated boundary conditions, for the RCS structural analysis.

2. Consideration of stresses resulting from flow induced vibrations due to pump operation and stresses in the pump casing resulting from the attachment of the restraints.

3. In Refs. 5 through 7, FPC proposed optimized restraint configurations consisting of combinations of snubbers and struts. In Refs. 5 and 6 the current groups of eight large bore snubbers per pump were replaced by groups of three restraints per pump, consisting of smaller snubbers or combinations of snubbers and struts. In Ref. 7, FPC further reduced the number of restraints per pump to two. The configuration formed by the two pin-connected restraints and the pump body is a structurally unstable configuration under compressive loading unless strongly restrained by the torsional stiffness of the piping system. No structural information was provided by FPC to enable us to make an independent assessment of the load carrying capacity of such a configuration. Furthermore, the maximum primary stress in the cold legs was shown to be 99% of the design allowable, while the maximum primary stress intensity in the pump casing was shown to be 94% of the allowable stress. Based on these facts, the concern arose that under actual loading the restraints would not experience the loads which the analysis indicated they would carry, and that the piping stresses and deformations would be higher than calculated. The analysis was performed on a linear elastic basis, while the actual stress state can be determined only from a non-linear structural analysis, which may, however, also indicate that a linear analysis is justified.

4. The optimization structural analysis specified modal damping values of Regulatory Guide (R.G.) 1.61 for RCS components and the interior concrete, and ASME Code Case N-411 for the piping segments. One concern was the method used for calculating equivalent or uniform modal damping values since structural analysis using the modal superposition method can specify only uniform modal damping values. Another concern regarding the damping values was the following: as stated above, the modal damping value for the pumps was taken as 2%, per R. G. 1.61. However, for these pumps a total of 32 large bore snubbers were replaced by a total of four smaller snubbers and four struts, thus reducing significantly the physical sources of

damping in the structure. Thus, the concern was that the damping specified by R.G. 1.61 for the pumps might be too high and some lower value should have been specified.

These concerns were discussed with FPC and B&W personnel during an NRC staff visit to the B&W offices in Lynchburg, Virginia, on February 27, 1986.

Concern No. 1 was addressed by an audit review of the initial design document for the RCS, B&W Report 32-1103808. This report was prepared by Franklin Research Labs (FRL) for B&W in 1970. The review was performed by the staff in the B&W office in Bethesda, Maryland, during the week of March 10, 1986. The staff reviewed the basis for the modeling of the RCS and the reactor building interior concrete and found the methodology acceptable. Therefore, this concern is considered resolved.

Concern No. 2 was discussed with B&W at the meeting in Lynchburg, Virginia, and a response submitted in Ref. 9. It was stated that flow-induced vibration due to pump operation results in stresses below 500 psi and that these stresses are negligible in the ASME Code fatigue analysis. In addition, there are no integral attachments to the pump casing; the restraints are attached to a heavy ring attached to the motor stand, which experiences low stresses. We have accepted these responses and consider this concern resolved.

To resolve Concern No. 3, B&W was requested to perform an analysis of a structural model consisting of a pump, the attached cold legs, and the proposed two-restraint configuration. This model was subjected to static loading such that the restraints would experience compression. The analysis was performed using the computer program ANSYS, which has the capability of performing non-linear structural analysis. The results are summarized in Ref. 9.

Based on this analysis, B&W demonstrated the following:

a. The extent of linear behavior of the system depends on the stiffness of the piping system. Reducing the stiffness significantly induces highly non-linear behavior, thus verifying that there was a basis for the concern.

b. The actual stiffness of the piping system is large enough such that in this case linear, stable behavior is assured with a two-restraint configuration.

We reviewed the results of this analysis and have found them acceptable. This concern is therefore considered resolved.

Concern No. 4 was also addressed in the review of B&W Report 32-1103808. Modal damping for structures which may have different damping in various substructures is determined by the calculation of composite modal damping values. The B&W procedures are in accordance with the methodology described in the ASME B&PV Code Section III, Appendix N, Section N-1233, which addresses the calculation of composite damping. We reviewed the procedures used by B&W and found them acceptable. This concern is therefore considered resolved.

In response to the second part of this concern, B&W stated at the meeting in Lynchburg that the modal damping values stated in R. G. 1.61 are applicable to equipment and structures regardless of the number of supports. Since there is no basis for requiring modal damping values lower than those specified in R.G. 1.61, other than direct testing which is not presently feasible, we have accepted the B&W position and consider this concern resolved.

Based on our review, we conclude that the plant-specific analysis is acceptable.

B. Vendor-Calculated Envelope

In addition to the reanalysis of the reactor coolant loop performed by B&W, FPC has also shown in Ref. 7 that the maximum bending moments resulting from the reconfiguration of the RC pump supports at CR-3 are within the envelope of moments that have been justified relative to the LBB Topical Report for B&W plants (Ref. 3). FPC has also stated that the safety factors associated with the new CR-3 loads are greater than the limiting factors reported in Ref. 3, and therefore since the LBB Topical Report envelopes CR-3, this plant is not a limiting plant with respect to LBB. We find these statements acceptable.

C. Reliability

The licensee has reduced the number of snubbers needed for restraining the RC pumps from 32 to four snubbers and four link-bar/struts. This has enhanced the system reliability under normal operating conditions by reducing the possibility of stresses in the piping and components which could be induced by inadvertent snubber lockup during plant heat-up/cooldown operation. In addition, the increased flexibility of the system will be able to distribute more efficiently the internal loading and stresses resulting from seismic loading, which for this plant has a low probability of occurring. FPC has stated that the new snubbers have various technical improvements which increase their individual operating reliability, and has also demonstrated that all heavy component supports have ample design margins with respect to applicable load ratings or allowable stresses. This constitutes a demonstration of overall system reliability which we find acceptable.

D. Independent Review

FPC chose the A/E firm Gilbert Associates to perform the independent design verification of the RC pump support optimization because of their familiarity with the plant layout and design. Gilbert was the original A/E on CR-3, but was not directly involved in the design of the RC piping and components, thus satisfying the independence criteria.

The efforts of the independent design verification program were discussed with FPC, B&W and Gilbert personnel during the staff meeting at the B&W offices in Lynchburg, Virginia, on February 27, 1986, and were found to be acceptable.

E. Leak Detection Capability

By letter dated October 29, 1985 (Ref. 11), FPC provided a report entitled "Assessment of CR-3 RC Leak Detection System" which described the reactor coolant pressure boundary (RCPB) leak detection design. RCPB leak detection capability consists of the following methods:

- Reactor Building Sump Level and Flow
- Reactor Building Airborne Particulate Monitoring (Radioactivity)
- Reactor Building Airborne Gaseous Monitoring (Radioactivity)
- Makeup Tank Level
- Reactor Coolant System Water Inventory Balance
- Condenser Off-Gas Monitor
- Decay Heat Closed Cycle Cooling Liquid Radiation Monitors
- Nuclear Service Closed Cycle Cooling Liquid Radiation Monitors
- Reactor Building Pressure

The reactor building sump level, airborne particulate and airborne gaseous monitors each have the capability of detecting a one gpm change in leak rate in less than one hour, which exceeds the four hour maximum capability specified in NRC Generic Letter 84-04. Further, the existing CR-3 Technical Specifications limit plant operation at power to 30 days when only two RCPB leakage detection systems are operable. The Technical Specifications for leakage detection systems have a sensitivity for detection of a 1 gpm leak within one hour. These Technical Specifications satisfy the NRC staff criteria for assuring availability and operability of the necessary leakage detection measures.

Based on the above, we conclude that FPC's RCPB leak detection system capability satisfies the NRC staff criteria of Generic Letter 84-04 and therefore that appropriate leakage detection devices are installed and that criteria for their availability and operability have been met.

Based on our evaluation as summarized above, we conclude that FPC has satisfied the guidance accompanying the modified GDC-4 and therefore may exclude from the design basis for the reactor coolant pump supports the dynamic effects associated with postulated ruptures of primary coolant loop piping and may modify the reactor coolant pump supports as described in the references herein.

FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

State Consultation

In accordance with the Commission's regulations, consultation was held with the State of Florida by telephone. The State had no comment on this proposed amendment.

Response to Comments

No comments were received in response to the Federal Register Notice of May 7, 1986 (51 FR 16942).

No Significant Hazards Consideration Determination

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a proposed license amendment involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The amended GDC-4 states that "the dynamic effects associated with postulated pipe ruptures of primary coolant loop piping in pressurized water reactors may be excluded from the design basis when analyses demonstrate the probability of rupturing such piping is extremely low under design basis conditions." Based on the licensee's submittals and the Commission's review of these submittals, the advanced fracture mechanics techniques employed provide assurance that flaws in primary system piping will be detected before they reach a size that could lead to unstable crack growth. Therefore, the probability of large pipe breaks in the primary coolant system is sufficiently low such that dynamic effects associated with postulated pipe breaks need not be a design basis. In addition, the revised design for the reactor coolant pump supports adequately considers all remaining design basis loads. The accident mitigation features (e.g., emergency core cooling system, containment) of the plant are not affected by the proposed amendment. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Based on the Commission's review, the revised design for the reactor coolant pump supports adequately considers all remaining design basis loads. The proposed change introduces no new mode of plant operation. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Based on the Commission's review of the licensee's submittals, Code design criteria for the reactor coolant piping will not be exceeded. The revised design of the reactor coolant pump supports will result in improved support system reliability and may therefore improve overall safety margins for the plant. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. Thus we conclude that:

(1) Operation of the facility in accordance with the proposed amendment would not significantly increase the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Therefore, we conclude that this amendment involves no significant hazards considerations.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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 made a final no significant hazards consideration finding with respect to this amendment. Accordingly, this 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 this amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 23, 1986

Principal Contributors: M. Hartzman, J. Wermiel

References:

1. Letter 3F0285-02, G. R. Westafer to H. R. Denton, "Request for Exemption From a Portion of 10 CFR 50, Appendix A, General Design Criterion 4" dated February 1, 1985.

2. Generic Letter 84-04, D. G. Eisenhut to PWR Licensees, Construction Permit Holders and Applicants for Construction Permits, dated February 1, 1984.

3. Babcock & Wilcox Owners Group Report, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," B&W Topical Report BAW-1847, dated September 1984.

4 NRC letter, D. M. Crutchfield to L. C. Oakes of the B&W Owners Group, subject "Safety Evaluation of B&W Owners Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," dated December 12, 1985.

5. Letter 3F0885-24, G. R. Westafer to H. R. Denton, "Reevaluation of CR-3 Cooling System Loads Utilizing Leak-Before-Break Concept to Remove Reactor Coolant System Main Loop Pipe Break Protective Devices," dated August 30, 1985.

6. Presentation by FPC at the second meeting between NRC/FPC to discuss partial exemption from GDC-4, October 31, 1985.

7. Letter 3F0186-12, G. R. Westafer to H. R. Denton, "Additional Information Regarding Request for Partial Exemption from General Design Criterion 4" dated January 13, 1986.

8. Letter 3F0186-22, E. C. Simpson to H. R. Denton, "Snubber Optimization Request" dated January 21, 1986.

9. Letter 3F0486-02, G. R. Westafer to H. R. Denton, "Snubber Optimization Approval Request," dated April 2, 1986.

10. Letter 3F0486-13, G. R. Westafer to H. R. Denton, "Proposed License Amendment-GDC-4, dated April 24, 1986.

11. Letter 3F1085-13, G. R. Westafer to H. R. Denton, "Assessment of CR-3 RC Leak Detection System, dated October 29, 1985.