January 28, 1977

SUPPLEMENT NO. 4

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

IN THE MATTER OF

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT NO. 3

DOCKET NO. 50-302

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1.0 INTRODUCTION

The Nuclear Regulatory Commission's (Commission) Safety Evaluation Report on the matter of the application by Florida Power Corporation, et al to operate the Crystal River Unit No. 3 facility was issued on July 5, 1974. Supplement No. 1 to our Safety Evaluation Report was issued on January 13, 1975. Supplement No. 2 was issued on December 3, 1976, and Supplement No. 3 was issued on December 30, 1976.

We concluded in Supplement No. 2 to our Safety Evaluation Report that the Crystal River Unit No. 3 facility may load fuel and be operated in the refueling mode of operation and the cold shutdown mode of operation (modes 5 and 6 as defined in the plant Technical Specifications). On December 3, 1976 the Commission issued Facility Operating License No. DPR-72 to Florida Power Corporation and eleven co-owners (licensees) authorizing operation of the facility in the refueling mode of operation and the cold shutdown mode of operation.

In Supplement No. 2 to our Safety Evaluation Report we stated that upon satisfactory resolution of a number of outstanding safety items, power operation may be authorized. In Supplement No. 3 to our Safety Evaluation Report we concluded that all of the outstanding safety items had been resolved to the extent that plant operation is acceptable within the limitations discussed in Sections 3.8.1 and 6.3.3 of Supplement No. 3.

On December 30, 1976 the Commission issued Amendment No. 1 to the facility operating license. The amended license authorizes plant operation at power levels not to exceed five percent of rated power (startup mode 2 as defined in the plant Technical Specifications). This limitation on plant operation would be removed when the Commission has completed its review of the structural integrity test of the containment and confirmed its conclusion that the repaired structure meets the original structural design criteria.

Amendment No. 1 to the facility operating license further restricted plant operation to not more than 82 percent of rated thermal power as limited by the revised plant Technical Specifications, issued as an attachment to Amendment No. 1, which placed limiting conditions of operation on the regulating rod group insertion limits and axial power imbalance until we have satisfactorily completed our review of additional information submitted by Babcock and Wilcox relative to its evaluation model of the emergency core cooling system.

As discussed in Section 3.8.1 of this Supplement, we have completed our review of the structural integrity test of the containment and have confirmed our conclusion that the repaired structure meets the original design criteria and is acceptable.

As discussed in Section 6.3.3 of this Supplement, we have satisfactorily completed our review of the Babcock and Wilcox evaluation model of the emergency core cooling system.

Accordingly we conclude that all of the limitations on plant operation contained in Amendment 1 to the license may be removed and that operation of the facility at full rated power is acceptable.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.8 Design of Category I (Seismic) Structures

3.8.1 Containment

We stated in Supplement No. 3 to our Safety Evaluation Report that based on our review to date we concluded that the plant can be operated within the startup mode of operation at power levels less than five percent of rated thermal power without adversely affecting the health and safety of the public. This limitation was placed on plant operation until we completed our review of the structural integrity test of the containment to confirm our conclusion that the repaired structure meets the original design criteria.

We also stated that we would require Florida Power Corporation to propose additional surveillance requirements on the containment in order to provide assurance that the structure will continue to behave as predicted during the life of the plant, and that our principal concern in this regard is the strains that may be introduced as a consequence of temperature differentials across the containment dome. Accordingly, we conditioned the facility operating license in Amendment No. 1 to the license to require that the additional surveillance program be submitted to the Commission within three months of the date of issuance of the license.

As we stated in Supplement No. 2 to our Safety Evaluation Report, we required Florida Power Corporation to make a detailed analysis of the repaired dome and to have it instrumented so that a correlation between the predicted and measured behavior can be established when the containment is subjected to the structural integrity test conducted according to the recommendations of Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Containments."

We have completed our review of the final test report, and have compared the measured dome strains to the predicted values. We have also compared the measured displacements of the dome during the structural test with the predicted displacements.

Comparisons of the measured to the predicted strains and displacements of the containment dome indicate that the structure behaved in an acceptable manner. The structural integrity test therefore confirms our conclusion that the structure meets the original design criteria and will withstand the specified design conditions without impairment of structural integrity or safety function.

Based on the determinations indicated above, we conclude that with regard to the containment structural design, and repair of the containment dome, operation of the facility at full rated power is acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

In Supplement No. 3 to our Safety Evaluation Report we stated that Babcock and Wilcox was considering several approaches to resolving the matter with regard to its evaluation model which allows a return to nucleate boiling after critical heat flux conditions have been reached during the blowdown phase of a postulated loss-of-coolant accident. Until this matter was satisfactorily resolved, we conditioned the operating license to limit plant operation to 82 percent of rated power. The plant Technical Specifications were revised to restrict the regulating rod insertion limits and power imbalance in order to reduce the linear heat generation rate by 20 percent of the value used in the performance evaluation of the emergency core cooling system.

We stated in Supplement No. 3 to our Safety Evaluation Report that we concluded that power operation of the facility was acceptable provided that the linear heat generation rate of the fuel elements is reduced by 20 percent of the values used in the performance evaluation of the emergency core cooling system. This limitation on the linear heat generation rate is achieved by limiting the regulating rod insertion limits and the power imbalance according to revised plant Technical Specifications that were issued with Amendment No. 1 to the facility operating license on December 30, 1976.

Babcock and Wilcox submitted a proposed revision to their evaluation model to preclude a return to nucleate boiling in its letter of December 20, 1976, from K. E. Suhrke to S. A. Varga (NRC). After a meeting with the staff, on January 21, 1977, Babcock and Wilcox submitted a further revision to their proposed model in response to staff concerns. This revision is described in the attachment to its letter of January 24, 1977, and will be incorporated into Topical Report B&W-10104A, Rev. 1, "B&W's ECCS Evaluation Model." The staff has reviewed the proposed modification and concludes that the blowdown heat transfer logic results in a model which is consistent with the post-critical heat flux heat transfer requirements of paragraph I.C.5 of Appendix K to 10 CFR Part 50. Sensitivity calculations indicate that the overall effect of the model change on peak clad temperature is less than 20°F.

Babcock and Wilcox has provided revised emergency core cooling system performance calculations for the worst case break using the revised evaluation model which demonstrates that the peak clad temperature and the percent of local and core-wide metal-water reaction remain below the limits specified in Section 50.46 of 10 CFR Part 50. The reanalysis shows that the deletion of the return to nucleate boiling has no effect on the loss-of-coolant accident limits for the worst case break. Consequently, recomputations for other breaks would not affect the size or location of the worst case break.

We have completed our review of the additional analyses using the small break model and the transition break analysis using both the small break model and the large break model which were discussed in Supplement No. 3 to our Safety Evaluation Report. The results of these analyses confirm that the worst case break is the 8.55 square foot, double-ended break analyzed in Topical Report BAW-10103, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS." Based on this review, we conclude that the referenced break spectrum is acceptable.

We therefore conclude that the analysis of the emergency core cooling system performance conforms to the acceptance criteria in Section 50.46 of 10 CFR Part 50 in accordance with methods described in Appendix K to 10 CFR Part 50.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth in our Safety Evaluation Report issued on July 5, 1974, and Supplement Numbers 1, 2 and 3 to that report issued on January 13, 1975, December 3, 1976, and December 30, 1976, respectively, and our evaluation as set forth in this Supplement, we conclude that the facility operating license may be amended to allow power operation at full rated power.

We conclude that the activities authorized by the amended license can be conducted without endangering the health and safety of the public, and we reaffirm our conclusions as otherwise stated in our Safety Evaluation Report and its Supplements.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW
CRYSTAL RIVER UNIT NO. 3

271. December 23, 1976	Letter from Florida Power Corporation transmitted corrected data related to the structural integrity test of the containment.
272. December 30, 1976	Supplement No. 3 to the Safety Evaluation Report issued.
273. December 30, 1976	Amendment No. 1 to Facility Operating License No. DPR-72 issued.
274. January 7, 1977	Letter to Florida Power Corporation regarding reactor pressure vessel overpressurization.
275. January 12, 1977	Letter to Florida Power Corporation regarding reactor coolant system flow indication.
276. January 12, 1977	Letter to Florida Power Corporation authorizing plant operation in Startup Operational Mode 2.
277. December 16, 1976	Letter from Babcock and Wilcox related to its ECCS evaluation model.
278. December 21, 1976	Meeting with Babcock and Wilcox to discuss the ECCS evaluation model.
279. December 24, 1977	Letter from Babcock and Wilcox related to modifications to its ECCS evaluation model.

Note: Correct Item 257 to December 3, 1976.