Florida
Power
CORPORATION

September 27, 1985
3F0985-26

Mr. Harold R. Denton
Office of Nuclear Reactor Regulation
Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Transmittal of Report Related to Request for Exemption from a Portion of 10 CFR 50 Appendix A, General Design Criteria 4

Reference:
1) Florida Power Corporation (FPC) letter to NRC, Westafer to Denton, dated February 1, 1985 (3F0285-02), subject Request for Exemption from a Portion of 10 CFR 50, Appendix A, General Design Criteria 4 (GDC-4).
2) FPC letter to NRC, Westafer to Denton, dated August 30, 1985 (3F0885-24), subject Re-evaluation of CR-3 Reactor Cooling System Loads Utilizing Leak-Before-Break Concept to Remove Reactor Coolant System Main Loop Pipe Break Protective Devices.

Dear Sir:

The reference 1 letter requested an exemption from a portion of the GDC-4 requirements in order to utilize the Leak-Before-Break concept at Crystal River Unit 3 (CR-3) and presented a sequence of actions (tentative dates of reports) which would provide additional justifications for the reduction at CR-3 in the number of large bore hydraulic snubbers restraining the reactor coolant pumps.

The reference 2 letter provided the initial report submitted in the sequence shown in reference 1.

The report enclosed with this letter is the second to be submitted in the sequence shown in reference 1 and is an assessment of the benefits and risks of eliminating Loss-Of-Coolant-Accident protective devices currently used in the CR-3 nuclear electric generating plant.

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We are arranging with NRC staff a technical meeting with FPC and B&W representatives in late October, 1985 to discuss the status of the NRC review and to identify needs for additional information, if required. This meeting date is consistent with our continued need for informal input from NRC during November 1985 to permit procurement by FPC of an optimized snubber arrangement to replace the current design.

Sincerely,

E C Simpson

E. C. Simpson
Director, Nuclear Operations
Engineering & Licensing

EHD/feb

Enclosure: Report, B&W Document ID: 51-1159048-00, Safety Balance Assessment for Elimination of Reactor Coolant System Main Loop Break Protective Devices, Crystal River 3 Generating Plant