June 6, 2002

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing and Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 — REQUEST FOR ADDITIONAL INFORMATION RE: PROPOSED LICENSE AMENDMENT REQUEST NO. 263, REVISION 0, RELOCATION OF REACTOR COOLANT SYSTEM PARAMETERS TO THE CORE OPERATING LIMITS REPORT AND 20 PERCENT STEAM GENERATOR TUBE PLUGGING (TAC NO. MB2499)

Dear Mr. Young:

By letter dated July 24, 2001 (3F0701-11), you submitted an amendment application to revise the Crystal River Unit 3 (CR-3) Improved Technical Specifications Tables 3.3.1-1, 3.4.1, and 5.6.2.18 to relocate the reactor coolant system parameters to the Core Operating Limits Report and to increase the steam generator tube plugging limit to 20 percent. The U.S. Nuclear Regulatory Commission (NRC) staff is currently reviewing your request, and it has determined that additional information is needed. The enclosed Request for Additional Information (RAI) was previously discussed with your staff in May 23 and June 5, 2002, telephone conference calls.

The enclosed RAI numbers 1 - 9 pertain to responses provided by Florida Power Corporation (FPC) in its letter to the NRC dated June 5, 2002. The responses were to an RAI submitted by NRC to FPC dated December 27, 2001. RAI number 10 is a new issue. In the May 23, and June 5, 2002, telephone conference calls, CR-3 staff indicated the enclosed questions contain no information considered proprietary by Framatone Technologies or FPC.

For the staff to complete its review on schedule, your response to this RAI is requested by June 14, 2002. This date was mutually agreed upon in a telephone conference call with CR-3 personnel on May 23, 2002. If circumstances result in the need to revise the target date, please call me at the earliest opportunity at 301-415-1437.

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

Enclosure: Request for Additional Information

cc w/encl: See next page

<u>REQUEST FOR ADDITIONAL INFORMATION</u>

<u>RELOCATION OF REACTOR COOLANT SYSTEM PARAMETERS TO THE CORE OPERATING LIMITS REPORT AND 20 PERCENT STEAM GENERATOR TUBE PLUGGING</u>

<u>CRYSTAL RIVER UNIT 3</u>

1. RAI 1.d - FPC stated that experimental data from vessel model flow tests were used to correlate inlet nozzle flow asymmetry with core inlet velocity asymmetry. Identify and reference these specific test programs and provide justification for the applicability of the experimental data to CR-3.

2. RAI 1.e - FPC bases the Reactor Coolant System flow reduction analysis on a generic Babcock & Wilcox (B&W) Owners Group study performed to evaluate 20 percent tube plugging for B&W designed 177 Fuel Assembly plants. Is the generic study referred to in response to this question the same one referred to in responses to questions 4.d, 6 (for Station Blackout (SBO) and Anticipated Transient Without Scram), 10.a and 12.a? FPC relies on several analyses performed as part of this study for justification for CR-3 amendment request. Provide the reference to B&W 51-5009660-01 and describe the information contained and how the specific information pertains to CR-3. Identify if there have been any changes in the design of CR-3 since the time of this generic study which would invalidate the results, and provide adequate justification for continued acceptability from the results of the study?

3. RAI 4.b - Regarding the expected increase in clad oxidation levels, FPC states that the highest burnup pins are approaching the clad oxide limit but that acceptable results have been demonstrated by the use of pin-specific power histories. Provide a comparison of clad oxidation results assuming 0 percent and 20 percent tube plugging. Also, provide more detail regarding the methodology used to determine pin-specific power histories.

4. RAI 4.d - Provide a reference to the generic analysis used to demonstrate that no saturation occurs in the guide tube assembly hold-down springs, guide tubes and spacer grids. Provide the reference to B&W 51-5009660-01 and describe the information contained and how the specific information pertains to CR-3. Identify if there have been any changes in the design of CR-3 since the time of this generic study which would invalidate the results, and provide adequate justification for continued acceptability from the results of the study?

5. RAI 6 - For the Loss of Flow transients (Four and One Reactor Coolant Pump (RCP) Coastdown, Locked Pump Rotor) FPC states that a new system analysis was not required and only reanalyzed the Departure from Nucleate Boiling (DNB) portion of the event. Justify the assumption that the normalized core inlet flow profile does not change as a result of 20 percent tube plugging. Include an assessment of the impacts of a change in the inlet flow profile on the acceptance criteria for these events as listed in the CR-3 Updated Final Safety Analysis Report. Also, explain the difference between minimum design flow rate and minimum DNB flow rate, and specify the reduction in flow rate used in the analysis. It is not clear which flow rate curve was actually used in the

analyses because initially FPC states the minimum design flow rate is used, but later states the minimum DNB flow rate is reduced. This needs clarification.

6. RAI 6 - FPC states that the consequences of the loss of all alternating current (ac) power accident are bounded by the loss of main feedwater accident because the net energy addition to the primary coolant during the loss of ac power transient is less due to the RCP's tripping immediately upon a loss of power. Identify the specific consequences referenced? Also, although the energy addition to the primary coolant may be less for the loss of ac power transient, the RCP trip produces a countering effect due to loss of forced flow. Quantify the impact of the loss of forced flow on DNB Ratio due to the RCP trip.

7. RAI 6 - Discuss the impacts of 20 percent tube plugging on Condensate Storage Tank inventory required and available for the 4-hour coping period for an SBO event.

8. RAI 8 - Provide a quantitative assessment of 20 percent tube plugging on the Feedwater Line Break accident results. This is not addressed in either the original submittal or the RAI responses.

9. RAI 13 - FPC states that the new Improved Technical Specification minimum Reactor Coolant System (RCS) loop pressure limit of 2064 psig provides assurance that the nominal 2200 psia core exit pressure is maintained. FPC provided a discussion of the methodology used to correlate the 2064 psig measured pressure to the nominal 2200 psia core exit pressure. Additional detail is required:

    A. Provide the location of the RCS pressure instruments.

    B. Provide the values for the pressure measurement uncertainty and the conservative representation of the delta-P core exit to pressure tap. A brief discussion of the actual calculation methodology would be useful.

    C. Provide a more detailed basis for the change in the minimum RCS loop pressure limit from 2061 to 2064 psig. Include a detailed discussion of the calculations / computer programs used to determine this final value.

10. To show that the referenced generically approved LOCA analysis methodologies apply specifically to CR-3, the staff requests that FPC provide a statement that CR-3 and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters.

June 11, 2002

Mr. Dale E. Young, Vice President
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ATTN: Supervisor, Licensing and Regulatory Programs
15760 W. Power Line Street
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/RA/
John M. Goshen, Project Manager, Section 2
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosure: Request for Additional Information

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ADAMS ACCESSION NO. ML021620613 *See previous concurrence

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