August 1, 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. 270, REVISION 0, POWER UPRATE TO 2568 MWT (TAC NO. MB5289)

Dear Mr. Young:

By letter dated June 5, 2002 (3F0602-05), you submitted an amendment application to revise the Crystal River Unit 3 (CR-3) Improved Technical Specifications Section 1.1 and License Condition 2.C.(1) to allow operation up to 2568 MWt. The enclosed Request for Additional Information (RAI) was previously discussed with your staff in a July 24, 2002, telephone conference call.

For the staff to complete its review on schedule, your response to this RAI is requested no later than August 13, 2002. This date was mutually agreed upon in a telephone conversation with CR-3 personnel on July 24, 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

REQUEST FOR ADDITIONAL INFORMATION

POWER UPRATE TO 2568 MWT

CRYSTAL RIVER UNIT 3

1. As seen in Section 4.12.1 of its submittal dated June 5, 2002, Florida Power Corporation (FPC, the licensee) did not perform containment integrity analyses for Crystal River, Unit 3 . FPC stated in the submittal that the original calculations that provided the mass and energy release data for the reactor building pressure analysis was performed, but were not located. However, FPC did not explain in the submittal whether they have performed a long-term Loss of Coolant Accident (LOCA) analysis for the uprated power to determine the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA. The staff needs this information to perform its review. Please provide information for the reactor building integrity analysis. Discuss the assumptions, computer code used, and calculation results for the containment accident pressure responses at the stretch rated power level.

2. Section 4.8.1 of FPC's June 5, 2002, submittal states that the main steam system is not impacted by the uprate. FPC should explain whether the main steam system and its components to operate at 2568 Mwt were enveloped by the original design. Please provide information on maximum flow rate and steam pressure corresponding to the thermal power rate originally designed. Explain whether the safety relief valves are capable to support the operation at the stretch power conditions with increased flow rate.

3. Section 4.8.18 of FPC's June 5, 2002, submittal states that the nuclear services and decay heat seawater (RW) system provides cooling water to the nuclear services closed cycle cooling (SW) system and decay heat closed cycle cooling (DC) system for heat removal during accidents and normal operation. FPC indicated in the submittal that the power uprate has no impact on this system based on its accident analyses. However, at uprated power level, the plant heat discharges to the RW system during normal operation and after accidents increase. The RW system draws seawater from the Gulf of Mexico to cool these safety-related cooling systems. The seawater temperature in the summer for the heat sink is normally higher in the Gulf area. Therefore, at the stretch rated power level, the service water temperature in SW and DC systems will increase from current level. Please explain whether the temperature variation of heat sinks resulting from weather changes was considered in the accident analysis.

4. FPC did not evaluate the effects of the proposed stretch rated power on equipment qualification to meet 10 CFR 50.49 requirements. Since this is a license amendment for power uprate, the licensee is required to confirm that the temperature profile previously approved for equipment qualification bounds the environmental conditions resulting from LOCA or Main Steam Line Break at the higher power level.

5. In its June 5, 2002, submittal FPC stated that it used the following references in its evaluaton:

   5. FRA-ANP 86-1266133-01, "CR-3 PT Fluence Analysis Report - Cycles 7-10

   6. FRA-ANP 32-5013936-00, "Adjusted Reference Temperature for 32 EFPY for CR-3 Power Uprate"

   The staff requests these documents in order to complete the evaluation on the changes on pressure vessel fluence.

ENCLOSURE

August 1, 2002

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Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing and Regulatory Programs
15760 W. Power Line Street
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John M. Goshen, Project Manager, Section 2
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cc w/encl: See next page

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Mr. Dale E. Young
Florida Power Corporation

CRYSTAL RIVER UNIT NO. 3

cc:
Mr. R. Alexander Glenn
Associate General Counsel (MAC-BT15A)
Florida Power Corporation
P.O. Box 14042
St. Petersburg, Florida 33733-4042

Mr. Jon A. Franke
Plant General Manager
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Mr. Michael A. Schoppman
Framatome ANP
1911 North Ft. Myer Drive, Suite 705
Rosslyn, Virginia 22209

Mr. William A. Passetti, Chief
Department of Health
Bureau of Radiation Control
2020 Capital Circle, SE, Bin #C21
Tallahassee, Florida 32399-1741

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

Mr. Craig Fugate, Director
Division of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

Chairman
Board of County Commissioners
Citrus County
110 North Apopka Avenue
Inverness, Florida 34450-4245

Ms. Sherry L. Bernhoft
Manager Regulatory Affairs
Crystal River Nuclear Plant (NA2H)
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Mr. Daniel L. Roderick
Director Site Operations
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Senior Resident Inspector
Crystal River Unit 3
U.S. Nuclear Regulatory Commission
6745 N. Tallahassee Road
Crystal River, Florida 34428

Mr. Richard L. Warden
Manager Nuclear Assessment
Crystal River Nuclear Plant (NA2C)
15760 W. Power Line Street
Crystal River, Florida 34428-6708