UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 2, 2012

Mr. Jon A. Franke, Vice President
Crystal River Nuclear Plant (NA2C)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT – REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST (TAC NO. ME6527)

Dear Mr. Franke:

By letter dated June 15, 2011, as supplemented by letters dated July 5, 2011; August 11, 2011 (two letters); August 18 and 25, 2011; October 11 and 25, 2011; December 15, 2011 (two letters); December 21, 2011; January 5, 2012 (two letters); January 19, 2012 (two letters); and January 31, 2012; Florida Power Corporation, doing business as Progress Energy Florida, Inc., submitted a license amendment request for an extended power uprate to increase thermal power level from 2609 megawatts thermal (MWt) to 3014 MWt for Crystal River Unit 3 Nuclear Generating Plant.

The U.S. Nuclear Regulatory Commission staff is reviewing the submittal and has determined that additional information is required to complete its evaluation. This request was discussed with Mr. Phil Rosean of your staff on February 22, 2012; and it was agreed that a response to the enclosed request for additional information would be provided within 45 days from the date of this letter.

If you have any questions regarding this matter, I can be reached at 301-415-1564.

Sincerely,

Siva P. Lingam

Siva P. Lingam, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

REGARDING EXTENDED POWER UPRATE TO INCREASE THERMAL POWER LEVEL

FROM 2609 MEGAWATTS THERMAL TO 3014 MEGAWATTS THERMAL

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

By letter dated June 15, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112070659), as supplemented by letters dated July 5, 2011; August 11, 2011 (two letters); August 18 and 25, 2011; October 11 and 25, 2011; December 15, 2011 (two letters); December 21, 2011; January 5, 2012 (two letters); January 19, 2012 (two letters); and January 31, 2012 (ADAMS Accession Nos. ML112010674, ML11228A032, ML11234A051, ML11234A427, ML11242A140, ML112860156, ML113040176, ML11354A232, ML11354A233, ML11361A460, ML12011A035, ML12030A209, ML12024A300 ML12024A301, and ML120330114, respectively), Florida Power Corporation (the licensee), doing business as Progress Energy Florida, Inc., submitted a license amendment request (LAR) for an extended power uprate (EPU) to increase thermal power level from 2609 megawatts thermal (MWt) to 3014 MWt for Crystal River Unit 3 Nuclear Generating Plant (Crystal River 3 or CR-3). In order to complete its review of the above documents, the Nuclear Regulatory Commission (NRC) staff requests for additional information (RAI) originating from our Electrical Engineering Branch (EEEB), Mechanical and Civil Engineering Branch (EMCB), and Health Physics and Human Performance Branch (AHPB):

**EEEB RAIs**

1. In your original LAR dated June 15, 2011 (ADAMS Accession No. ML112070659), Attachment 5, Section 2.3.1.2, pages 2.3.1-4 and 2.3.1-5, "Detailed EPU Impacts by Zone," the licensee provided an environmental qualification (EQ) reclassification of the plant area due to projected post-EPU environmental conditions, which impacts Zone 17 in the Intermediate Building, and Zones 18, 23, 47, 60, 62, and 78 in the Auxiliary Building.

   Provide a list of the equipment/components that will be affected as a result of the new areas being designated as "harsh." Provide the post-EPU radiation levels in these reclassified zones and the radiation levels to which these equipment/components are qualified.

2. In the original LAR, Attachment 5, Section 2.3.1.2, page 2.3.1-5, "Summary of EPU Impact on Component Basis," the licensee discussed the proposed plant changes as a result of EPU, as referred to Appendix E, which add or modify equipment subject to EQ requirements. The licensee further stated, "These new components will be designed, procured, and installed in compliance with appropriate EQ requirements."

Enclosure

Provide a detailed list (preferably in a table) of new EQ equipment, which are referenced in the above paragraph. Provide a discussion of EQ (similar to the EQ discussion provided in Section 2.4 of Appendix E, page E-38, for atmospheric dump valves and accessories) for the following:

(i) new EQ motor operated valves (MOVs) and the new EQ transmitters for the low pressure injection cross-tie and hot leg injection design change,

(ii) new EQ MOV for the feedwater booster pump modification,

(iii) new EQ differential pressure indicating switches, and

(iv) EQ core exit thermocouple for the Inadequate Core Cooling Monitor System change.

3. In the original LAR, Attachment 5, Section 2.3.1.2, page 2.3.1-2, the licensee stated, "The environmental parameters evaluated for EPU conditions include temperature, pressure, radiation dose, submergence, chemical spray effects, and humidity, for both normal operation and post-accident conditions."

a) Confirm that the margins for the equipment qualified per guidance under Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974, "Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," are maintained under EPU conditions.

b) Original LAR Figure 2.3.1-1 (Post-EPU LOCA [loss-of-coolant accident] versus EQ Bounding Profile Comparison, Inside Containment Temperature), appears to show that the IEEE recommended temperature margin (15 °F) is not maintained at one point (1.00E+04 time/280 °F). Please clarify the apparent deviation.

c) Provide the post-EPU pressure profile for conditions inside containment to demonstrate that the integrated pressure curve is bounded by the current EQ bounding profile.

d) Provide the post-EPU temperature profile for main steam line break environmental conditions to demonstrate that the profile is less severe than LOCA as discussed on page 2.3.1-3, under "Summary EPU Impact" in Reactor Building.

4. In the original LAR, Attachment 5, Section 2.3.1.2, page 2.3.1-2, under "Description of Analyses and Evaluations," the licensee stated that "Even though the EPU design change packages are not yet issued for construction, EPU impacts on parameters important to EQ are known."

Provide clarification of this statement. Also, the NRC staff finds that in Attachment 10 of the original LAR, "List of Regulatory Commitments," the licensee states that CR-3 will implement all EPU modifications per Technical Report (TR) Attachment E. Confirm that

the licensee intended to refer to Appendix E, Major Plant Modifications, of Attachment 5 of the original LAR instead of TR Attachment E.

5. In the original LAR, Attachment 5, Section 2.3.1.2, page 2.3.1-4, the licensee stated, "The RB [Reactor Building] postulated maximum flood level is not impacted by the proposed EPU conditions."

   Indicate the maximum postulated flood level versus the levels of the equipment to demonstrate that the associated equipment is not impacted by the EPU conditions in the RB. Similarly, address the postulated flood level conditions in the Intermediate Building.

6. In the original LAR, Attachment 5, Appendix E, "Major Plant Modifications," Section 1.4.1, the licensee stated, "The power and control from the existing MOVs will be re-routed into the RB through two new electrical penetrations (converted mechanical penetrations 327 and 328 from the Triangle Room)."

   Confirm that the new electrical penetrations are environmentally qualified with margins for the installed life in accordance with the guidance provided in IEEE Standard-317, "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations."

7. In the original LAR, Attachment 5, Section 2.3.2.2 under Section 2.3.2, "Offsite Power System," the licensee discussed the adequacy of the switchyard main generator output breaker capacity. Also in Appendix E, the licensee stated that no changes to either switchyard (230 kV and 500 kV) are necessary to support EPU.

   Provide a discussion that demonstrates that other equipment such as breakers (other than the generator main breaker), disconnects, buses, current transformers, tie-lines, etc., have adequate capacity so that the existing switchyard is capable of supporting EPU conditions (i.e., the equipment have adequate margins between the maximum worst case steady-state load and the equipment ratings).

8. In the original LAR, Attachment 5, Section 2.3.3.2, page 2.3.3-3, the licensee mentioned upgrades to the isolated-phase bus duct and the pre- and post-EPU continuous current ratings.

   Provide a summary of the calculation used to determine that the calculated short-circuit current available at the bus duct is within the bus short circuit capacity rating.

9. In the original LAR, Attachment 5, Section 2.3.3.2, page 2.3.3-3, the licensee stated that the proposed Class 1E modifications would not result in any configuration changes that would adversely impact the maximum emergency diesel generator (EDG) loading currently assumed.

   Provide a discussion and a summary of the calculation that demonstrates that the EDG loading in the post-EPU state, after taking into account increased loads on the vital bus inverters and engineered safeguard bus (safety-related) instrument power systems, will remain within each EDG's capacity.

10. In the original LAR, Attachment 5, Section 2.3.3.2, page 2.3.3-4, the licensee stated, "All lower voltage buses, switchgear, and motor control centers were demonstrated to have sufficient voltages at the lowest operating voltage on the grid to assure operability of the connected equipment."

    Provide a discussion and a summary of the calculation that demonstrates that the degraded voltage relay and under voltage relay settings at the 480 volt load center buses are not adversely affected by operation under EPU conditions.

11. In the original LAR, Attachment 5, Section 2.3.3.2, page 2.3.3-2, the licensee stated, "The impact of the electrical load increase was evaluated using load flow, short circuit, and protection-coordination studies."

    Provide a summary of the evaluation/study that demonstrates that the short-circuit rating due to the increase in loads (Class 1E and Non-Class 1E switchgear buses) are not adversely impacted by the load increase.

12. In the original LAR, Attachment 5, Section 2.3.4.1, page 2.3.4-2, the licensee discussed the load impacts on the direct current (DC) power system due to the proposed EPU and stated that emergency feedwater flow increase/flow control modification has an impact on DC power but remains within the loading capabilities of the station batteries.

    Provide a comparison of the existing loads and the loads added to the DC power system as a result of the EPU. Also, provide the design rating for each safety-related and nonsafety-related battery at CR-3 to show that adequate capacity exists to support EPU conditions.

13. In the original LAR, Attachment 5, Section 2.3.5.2, the licensee addressed the systems and components necessary to cope with a station blackout (SBO).

    a) Provide a summary of SBO loads for both pre- and post-EPU conditions.

    b) Section 14.1.2.9.5.2 of the CR-3 Final Safety Analysis Report provides an analysis (based on NUMARC 87-00) that supports a coping duration of 4 hours for an SBO event at CR-3. This section indicates that the analyses were performed based on a core power level of 2772 megawatts thermal (MWt) and that no analyses are required to support a power uprate of 2609 MWt.

       Explain whether the analyses provided in the original LAR Section 2.3.5.2 were performed to support a power upgrade to an increased core power of 3014 MWt due to EPU conditions.

14. On January 31, 2012 (ADAMS Accession No. ML120330114), the licensee submitted supplemental information in Attachment C, "Clarification Information to the CR-3 EPU Technical Report Section 2.7.3.1 Regarding the [Fast Cooldown System] FCS Batteries." Based on its review of the supplemental information, the NRC staff requests the following:

a) Discuss the impact of the EPU on battery room temperature during normal operation, design basis events, and beyond design basis events (such as an SBO).

b) Discuss the impact of placing the FCS batteries in the battery rooms has on battery room temperature during normal operation, design basis events and beyond the design basis events (such as an SBO).

c) Discuss the installation details of the FCS batteries in the battery rooms.

d) Discuss the seismic design considerations for the FCS battery installation and the potential seismic impact on the other safety-related equipment/components in the battery rooms.

**EMCB RAIs**

15. The control rod drive mechanism (CRDM) analysis discussion on page 2.2.2.4-1 of Attachment 5 of the original license amendment request (LAR) indicates that the extended power uprate (EPU) operating temperature of 608.7 °F slightly exceeds the design analysis temperature of 608 °F. The discussion does not indicate that the licensee intends to revise the design report to reflect this change. Please provide assurances that the CRDM analysis either has been or will be revised to include a discussion of the new operating temperature. The revision needs to include a statement that Code requirements continue to be met under the new condition, which will be present at the proposed EPU power level.

16. The steam generator base support evaluation discussion on page 2.2.2.5-2 of Attachment 5 of the original LAR states that the base supports are designed to the 2000 Addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Subsection NF. However, neither Table 4-2 of the Crystal River Unit 3 (CR-3) final safety analysis report (FSAR) nor CR-3 FSAR Section 4.6.2.3 includes these criteria. The licensee needs to provide a reference to an existing reconciliation between the 2000 Addenda criteria and the criteria listed in the FSAR, or provide reconciliation between the two criteria.

17. The in-core instrumentation guide tubes evaluation discussion on page 2.2.6-2 of Attachment 5 of the original LAR indicates that no re-evaluation of thermal stresses is required based on the reactor T-cold decreasing by less than 1 °F. However, on page 2.2.2.3-1 of the LAR, the reactor T-hot temperature is listed as increasing by 6.6 °F with the EPU, giving an average reactor temperature increase of approximately 3 °F. State the impact of the increases of T-hot and T-average on the structural integrity of the incore guide tubes and provide a justification for not re-evaluating the stresses in these tubes based on apparently more complex thermal conditions at the proposed EPU power level.

**AHPB RAIs**

18. In Section 2.11.1.2, under "Description of Analyses and Evaluations," Question 2 CR-3 Response, 1., of the original LAR, it is stated, "The Loss of Subcooling Margin procedure will be revised to include specific guidance for ensuring each of the automatic actuation functions of the [Inadequate Core Cooling Mitigation System] ICCMS occur within their allotted time." Please clarify how the timing is monitored and how the start times are determined for each function.

19. In Section 2.11 of the original LAR, it is stated several times that Safety Parameter Display System (SPDS) and ICCMS perform the same functions independently, for example, "To support EPU the ICCMS and Safety Parameter Display System (SPDS) will independently perform the [High Pressure Injection] HPI flow monitoring function automatically," as stated in Section 2.11.1.2, under "Description of Analyses and Evaluations," Question 2 CR-3 Response, 3.

    a. Is the SPDS safety-related?

    b. Will the SPDS and the ICCMS be used by operators to verify each other's output?

    c. What guidance will be provided for situations when ICCMS and SPDS differ significantly?

    d. How can these instruments be independent when the SPDS HPI low range flow comes from HPI flow transmitters in ICCMS? Other examples are Reactor Coolant System (RCS) wide range pressure and RCS low range pressure.

    e. Additionally, Section 2.11.1.2, under "Description of Analyses and Evaluations," Question 4 CR-3 Response, of the original LAR contains the following statement: "The SPDS will provide backup indication to be used in conjunction with the ICCMS and will be available if the ICCMS based instrumentation was lost." How will operators be warned that the displays being obtained for SPDS from ICCMS are invalid?

20. Contrary to RS-001, "Review Standard for Extended Power Uprates," Section 2.11.1, "Human Factors," Question 5, of the original LAR, the licensee did not provide the implementation schedule for making the changes to the training program and the control room simulator. Provide the above information so that the NRC staff can determine whether the changes will be completed in a logical sequence and prior to operating under uprated conditions. If an implementation schedule is not available, the licensee should commit to completing the simulator changes prior to simulator training, and to completing all EPU-required training prior to operation under uprated conditions.

21. Were any human factors lessons learned from other plant EPU experiences? If yes, describe.

22. In Section 2.11.1.2, under "Introduction" the original LAR, it is stated, "When initiating a plant change, the engineering change (EC) process requires the completion of a Human

Factors review for changes that may impact the Control Room layout (alarms, indication, appearance or performance)." Provide a copy of a Human Factors review that was done for an EC supporting the EPU.

March 2, 2012

Mr. Jon A. Franke, Vice President
Crystal River Nuclear Plant (NA2C)
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15760 W. Power Line Street
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Siva P. Lingam, Project Manager
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