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

February 8, 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; and January 5, 2012 (two letters); 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 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. Dan Westcott of your staff on January 12, 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; and January 5, 2012 (two letters) (ADAMS Accession Nos. ML112010674, ML11228A032, ML11234A051, ML11234A427, ML11242A140, ML112860156, ML113040176, ML11354A232, ML11354A233, ML11361A460, ML12011A035, and ML12030A209, 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 Probabilistic Risk Assessment (PRA) Licensing Branch (APLA) and Instrumentation and Controls Branch (EICB):

APLA RAIs

1. In your August 11, 2011, response (ADAMS Accession No. ML1123A051), Appendix 1, ASME 2007 self assessment, observation FnO-SC-B1-2 indicates that the peer reviewer identified an incongruity in the amount of time operators have to initiate feed-and-bleed after a small break loss-of-coolant accident (SBLOCA). The EPU resolution includes an analysis for transient-initiated accident scenarios when reactor coolant pumps (RCP) are not tripped as well as a station blackout (SBO) analysis. The resolution does not indicate the maximum amount of time available for operators to initiate feed and bleed cooling for SBLOCA post-EPU. The resolution implies that feed and bleed cooling must be performed within 30 minutes; however, the SBO analysis suggests 60 minutes. Please verify the amount of time available for operators to perform feed and bleed cooling following SBLOCA for pre-EPU and post-EPU; and provide an explanation for differences or similarities in pre-EPU and post-EPU results.

2. Please verify that all outstanding issues from the individual plant examinations (IPE) and individual plant external events examinations (IPEEE) highlighted in the NRC safety evaluation have been satisfied in subsequent PRA updates. If there are outstanding items from the IPE and IPEEE, explain the impact of the proposed EPU for those particular issues.

3. Per NRC RS-001, "Review Standard for Extended Power Uprates" (ADAMS Accession No. ML033640024), the licensee is required to provide findings of any industry or independent peer review. Please explain the focus of the 2009 peer review.

4. Section 2.13.1 of the original license amendment request (LAR) dated June 15, 2011 (ADAMS Accession No. ML112070659), under section titled, "CR-3 Current Licensing Basis," paragraph four, states, "The current PSA [probabilistic safety analysis] model of record is based on internal events and internal flooding only." However, the introduction in Section 2.13.2 lists an at-power PSA model that includes "Fire." Please clarify this discrepancy.

5. The EPU may impact the frequency of a stuck open pressurizer power-operated relief valve (PORV) or safety relief valve, thus impacting the consequential loss-of-coolant accident (LOCA) frequency. In addition, pressurizer level can have larger variations due to the EPU. Please perform a sensitivity study that increases the PORV challenged frequency and provide the resulting delta risk metrics.

6. Installation of the low pressure injection (LPI) cross-tie and the boron precipitation line result in an additional pathway with the potential for an interfacing systems LOCA (ISLOCA). Therefore the ISLOCA frequency was increased by seven percent. Please provide the basis for determining a 7 percent increase and perform a sensitivity study that shows the impact on core damage frequency and large early release frequency for a larger frequency.

7. Installation of the fast cooldown system (FCS) introduces an additional avenue for inadvertently opening the atmospheric dump valves (ADVs). Provide an explanation for why the value of the secondary line break and spurious actuation frequency remain the same between pre-EPU and post-EPU with the addition of FCS and inadequate core cooling mitigation (ICCM) systems.

8. The following questions address excessive feedwater transient:

    A. In Section 2.13.2 of the original LAR under subject, "Excessive Feedwater," the licensee states, "The spurious actuation of the new ICCM system could impact emergency feedwater but has no impact on ICS [integrated control system] or the MFW [main feedwater] System." Also in Section 1.2.7 of Appendix E of the original LAR, the licensee states that emergency feedwater requirements will be raised from 275 gallons per minute (gpm) to 330 gpm to satisfy increase in decay heat. The licensee is requested to explain why the addition of ICCM increases the actual excessive feedwater initiating event frequency and why the licensee chooses not to model this frequency change. Please provide a sensitivity study showing the effect on risk metrics.

    B. If there were a transient that involved an increase in feedwater flow to the steam generator (SG), the SG water level may exceed the aspirator port level, thus preventing the preheating of the feedwater. If the increased thermal stresses on the tubes or shell wall are excessive, an SG tube rupture or steam generator shell failure accident could occur creating a LOCA. The initiating transient of concern

involves an increase in feedwater flow. According to Appendix E of the original LAR, replacement generators were originally specified and designed to 3010 MWt and were reevaluated to EPU conditions at 3030 MWt, thereby raising additional concerns on the adequacy of the generators to address this transient. The NRC staff requests sensitivity studies for the conditional probability of failure given a feedwater overfeed event based on updated excessive feedwater frequencies for post-EPU, and current excessive feedwater frequencies for pre-EPU. In addition, characterize the impact on operator actions for excessive feedwater for post- and pre-EPU conditions.

9. In Section 1.3.1 of Appendix E of the original LAR, the licensee notes that the decay heat removal system performs LPI functions and supports active boron precipitation controls. However, at EPU conditions, these two scenarios are not adequately supported by the current system design and performance. Therefore, the licensee plans to modify the system by cross-connecting the two trains inside the reactor building. Please explain how the licensee plans to address the decrease in defense-in-depth associated with this modification and provide additional detail on how success criteria was changed in the PRA to reflect two out of two requirements instead of one out of two requirements.

10. In Section 2.13.2 of the original LAR under label, "Emergency Feedwater and Auxiliary Feedwater (PSA System Model EF)," the licensee notes that the EPU will require changes to ensure that the emergency feedwater (EFW) pumps can deliver the required rated flow to the steam generators. However, PRA best-estimate analyses performed for the EPU show that additional flow is not required to prevent core damage. The licensee is requested to provide additional detail that summarizes how the conclusion for not changing success criteria was reached. Additionally, under label, "Fast Cooldown System," the licensee also establishes that plant-specific PRA analysis does not require the FCS system for medium break LOCAs. Please provide the NRC staff a summary of the delta between the design basis analysis and the plant-specific PRA analysis for the success criteria determination of the FCS system.

11. EPU associated conditions may reduce the time to reach boron solubility limits in the core for medium and large LOCAs. This condition can result in boron precipitation on the fuel assemblies, which reduces heat transfer rates, and may lead to core damage. The potential impact on risk is from an increase in probability of the operators failing to initiate safety injection within this time period. Please characterize how this issue is addressed for CR-3 and any associated risk implications.

12. Of 70 unique post-initiator operator actions developed for the CR-3 PRA, the licensee states that only 7 were impacted by the EPU. Based on previous EPU precedent, the NRC staff finds the number of human error probabilities (HEPs) impacted by the EPU to be extremely low. The licensee is requested to provide a change in HEP assessment from pre- to post-EPU for all operator actions impacted by the EPU that are either under 30 minutes; have Fussel Vessly (FV) > .005; or Risk Achievement Worth >2.

13. Please describe any new credited operator actions as a result of the EPU. (i.e., new operator action to lock out the ADVs actuation in event of fire).

14. The human reliability analysis (HRA) assumes a reduction in system time window from 60 to 55 minutes (approximately 9 percent). This is less than the approximately 15 percent increase in power sought by the EPU. Please describe the analysis for how the system time window is uniformly reduced for all impacted HEPs by approximately 9 percent.

15. Section 2.13.2 of the original LAR under label, "HRA Dependency Evaluation", notes that none of the timing changes expected for post-EPU conditions affected the dependency level. The following sentence depicts that some of the dependent HEPs were updated and included in the quantification process. Please provide the pre- and post-EPU dependent HEPs that changed; similar to Table 2.13-1 of the original LAR.

16. Table 2.13-1 of the original LAR provides the base and updated values for impacted human error probabilities. The NRC staff is requesting the licensee to review their analyses and explain why a 5-minute reduction in system time (approximately 9 percent) increases the operator failure probability in some cases by over 115 percent. Also, basic event QHUEFW9Y requires the operators to complete the action within a 10-minute time window. How does that correspond to the 60-minute base value and 55-minute updated value?

17. Due to increased decay heat during EPU operations, additional PORVs may be required for successful feed-and-bleed, especially if charging is unavailable. Describe changes in success criteria for successful feed-and-bleed pre-EPU and post-EPU, and the resulting risk implications.

18. The licensee has not performed a probabilistic seismic analysis for the plant, therefore, please verify that all structural plant modifications and anchoring of all replacement components for EPU will have the same or greater seismic capability than the current design basis.

19. Table 2.13-2 describes risk results without risk reduction modifications. Briefly describe all risk reduction modifications implemented as part of the EPU. If available, the NRC staff requests a table showing risk results with risk reduction modifications.

20. Describe how the EPU affects the ability of the operator to close containment during an outage in the event of loss of shutdown cooling. Please provide any changes to the existing guidance/procedures.

EICB RAIs

21. In response to EICB acceptance review RAI question 3, the licensee submitted Enclosure 3 "IEEE [Institute of Electrical and Electronics Engineers] 603-1991 and IEEE 279-1971 Compliance Matrix" on August 18, 2011 (ADAMS Accession No. ML11234A427), which provides a summary of how the inadequate core cooling monitoring system (ICCMS), FCS, ADVs and the affected portions of emergency feedwater initiation and control (EFIC) system will meet applicable clauses of IEEE 603-1991 and IEEE 279-1971. The licensee lists the system specification statements for each requirement criterion of IEEE 603-1991 and IEEE 279-1971, but

does not demonstrate (e.g., through the analysis or test) how the equipment meets the requirements in IEEE 603-1991 and IEEE 279-1971.

a. Please provide the failure modes and effects analyses, overall availability reports, reliability reports, the summary of factory acceptance test results, and additional information for those systems (ICCMS, FCS, ADVs and the affected portions of EFIC system) to allow the NRC staff to confirm that tests are conducted to demonstrate that the safety system performance is adequate to ensure completion of protection over the range of transient and steady-state conditions and meet the requirements in IEEE 603-1991.

b. Provide the detail summary of test results of ICCMS for Class 1E equipment per Regulatory Guide (RG) 1.89; seismic qualification per RG 1.100; electromagnetic and radio-frequency interferences qualification per RG 1.180; and qualified isolation used between the nonsafety-related RCP trip circuits and ICCMS per RG 1.75.

22. In the last paragraph of page 2.4.2.2-2 of the original license amendment request (LAR) dated June 15, 2011 (ADAMS Accession No. ML112070659), the licensee discussed the modifications of safety-related EFW recirculation flow control and the replacement of ADVs.

    Please describe how FCS, EFW recirculation flow control, and the new ADVs meet the requirement criterion of IEEE 603-1991 quality assurance (e.g., Sections 5.3 Quality, 5.4 Equipment Qualification, 5.15 Reliability) and provide the supporting documents.

23. In the last paragraph of page 2.4.2.2-2 of the original LAR, the licensee stated, "The EPU requires an increase in minimum required EFW flow and a decrease in maximum EFW actuation delay time."

    Provide the values of the original and revised EFW actuation delay times and demonstrate by the calculation or the design that the revised actuation delay time has been properly implemented.

24. In the second to the last paragraph of section "Analog Inadequate Core Cooling Mitigation System" (page Appendix E-48) of Attachment 5 of the original LAR, the licensee briefly described the design fail to a safe state of ICCMS.

    Please list the power supply for each initiation channel and each actuation train and explain in more detail how the ICCMS complies with the regulatory guideline in NUREG-0800, Appendix 7.1-C, Section 5.5 for design fail to a safe state.

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Siva P. Lingam, Project Manager
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