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. Dan Westcott of your staff on February 27, 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 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 Containment and Ventilation Branch (SCVB). The section and tables referenced in the following RAIs are from Attachment 5 to the original EPU application dated June 15, 2011.

SCVB RAIs

SCVB-1.1 The concrete surface area (105,941 ft^2) listed in Table 2.6.1-5, "Containment Structural Heat Sink Input," differs from the CR-3's containment concrete surface area as listed in Updated Final Safety Analysis Report (UFSAR) Table 14.45 (117,800 ft^2). Please explain the inconsistency between the two.

SCVB-1.2 As stated in Section 2.6.1.2, the Improved Technical Specification Limiting Condition for Operation 3.6.4, maximum value for containment pressure during normal operation, is being revised from 17.7 psia (3 psig) to 16.2 psia (1.5 psig) as a result of the EPU. The change was implemented in the containment accident analysis for short term loss-of-coolant accident (LOCA). However, higher initial containment pressure was assumed for long term LOCA and main steam line break analyses. Explain the reasons for this inconsistency?

SCVB-1.3 The three postulated single case failures are described in page 2.6.1-7 of the CR-3 EPU Technical Report. Please discuss the difference between the first single failure scenario (loss of offsite power (LOOP) with failure of one emergency diesel generator (EDG)) and the second single failure scenario (One Reactor Building (RB) spray pump fails to start with or without LOOP). The information is requested because it appears that a failure of one RB spray pump would be automatically covered by the LOOP with failure of one EDG.

Enclosure

SCVB-1.4 Section 2.6.3.2 provides the details of the main steam line break analysis at the EPU conditions. Explain the differences between the current licensing basis analysis and the EPU analysis, with special attention to the hardware modifications as a result of the EPU (e.g., modification of main feedwater (MFW) and MFW booster pumps). In particular, discuss all changes to the inputs, assumptions, single failures, MFW flow rates, MFW pump start times, and the codes used in the analysis. In addition, provide the reasons for considering a closure time of 31 seconds for the MFW isolation valves when faster closing isolation valves capable of closing in 21 seconds are being implemented for the EPU. Also, explain how feedwater flow from the MFW pump into the containment is apportioned through the MFW isolation valve during its closure?

SCVB-1.5 Please provide the EPU impact on the emergency core cooling system (ECCS) ability to provide homogeneous atmospheric mixing within containment. In accordance with the requirements of Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.44, Subsection (b)(1) as related to mixed atmosphere for currently licensed reactors, confirm that the CR-3 containment has the capability of ensuring a mixed atmosphere following a LOCA at EPU conditions. Summarize the CR-3's containment design that supports this assessment.

SCVB-1.6 The applicability of NRC Generic Letter (GL) 96-06 as it relates to CR-3 was addressed in Section 2.5.4.3, "Reactor Auxiliary Closed Cycle Cooling Water Systems." It was stated in this section that CR-3 implementation of the requirements of GL 96-06 was previously evaluated. Discuss how the previous evaluation regarding fluid contained in penetrations between containment isolation valves is affected (thermally induced overpressurization) and if any additional measures are required as a result of the EPU.

SCVB-1.7 Please discuss if the feedwater into and steam out of steam generator, and the steam generator metal in contact with secondary side fluid were considered when determining the sources of energy addition to containment on the mass and energy release analyses described in Section 2.6.3.2.

<u>Control Room Habitability and Ventilation Systems</u>

SCVB-2.1 Section 2.7.3, "Ventilation Systems," Subsection 2.7.3.1.2 discusses the ability of the Control Room Area Ventilation System (CRAVS) to maintain a mild temperature environment for control room personnel and control room components. Specifically, CR-3 evaluated the safety-related portions of the CRAVS (Control Complex Ventilation System and Emergency Feedwater Initiation and Control System). It was stated that the "heat load increases for EPU are small." Please provide a summary of the equipment changes in the control room that have an impact on heat loads, however small the impact may be. Specify if the heat load evaluations performed are qualitative or quantitative, and if qualitative, provide a basis for your conclusion.

SCVB-2.2 It is stated in Section 2.7.4, "Spent Fuel Pool Area Ventilation System," Subsection 2.7.4.2 that the air temperature in the spent fuel pool area is affected by heat released from the spent fuel pool. However, it is not clear how the heat load increase to the ventilation system is considered in the EPU evaluations. It was also stated in Subsection 2.7.5.2 that a very large temperature range (55 °F to 122 °F) is acceptable within the fuel handling area. Discuss the present systems margin in maintaining this temperature range, and how the additional heat load due to the EPU is evaluated to be within the margin.

SCVB-2.3 Section 2.7.5, "Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems," Subsection 2.7.5.2 discusses significant plant modifications that have a potential to significantly add to the heat load in the Turbine Building. Please explain if any calculations were performed to quantify the overall effect of the heat load additions on the Turbine Areas Ventilation System and the conclusions of the calculations.

SCVB-2.4 Section 2.7.7, "Reactor Building Ventilation Systems," Subsection 2.7.7.2 discusses the Reactor Building Recirculation System's function to control containment temperature via the Industrial Cooling System (CI). The licensee further states that CR-3 has developed procedures to shift from CI to two trains of Nuclear Services Closed Cooling Water (SW) for cooling during the challenging summer month periods. However, it is not clear if this shifting procedure is part of the current licensing basis, or if it will be new due to the EPU. Please provide additional details about the procedures. For instance, is the shifting automatic or manual? How does it affect the containment isolation function of these systems?

Based on UFSAR Section 9.7.2.1, the CI System provides chilled water to the RB recirculation system coolers. Normally, you would expect chilled water to be at a lower temperature than SW. Please explain how additional cooling is achieved through these coolers by shifting from chilled water to SW.

In Subsection 2.7.7.2, the licensee discusses the increased load in the RB. The licensee stated, "The ΔT across the hot-leg insulation increases by 6.4 °F (˜1.5%). Since the ΔT across the pressurizer insulation is unchanged and the ΔT across the cold-leg insulation is actually decreasing, the total heat loss from the RCS [Reactor Coolant System] will increase by less than 1.5%." The NRC staff would like to know if the additional heat loads in the RB also include the EPU related increases in the steam generator heat loads and the control rod drive (CRD) mechanism heat loads.

In the same subsection, the licensee discusses the temperature limit for the CRD shroud, which is 150 °F. Based on this limit, the electrical connector and CRD position indicator enclosures located in the service structure has been challenged during the summer months. It is further stated that qualified component lifetime is trended with the cumulative impact monitored and preventive maintenance actions implemented as appropriate. Is this presently done, or will this be initiated as a result of the EPU? Are these components covered under the Equipment Qualification program?

SCVB-2.5 Section 2.3.5, "Station Blackout," Subsection 2.3.5.2 concludes that the EPU will not affect the ability to fulfill the requirements of CR-3's Ventilation system during a station blackout event. It is stated in this section that the temperatures have been evaluated for the added EPU heat loads and found acceptable. Please provide the details of the evaluations performed, and compare the results with the pre-EPU conditions.

ECCS Pump Net Positive Suction Head

The issue of crediting containment accident pressure (CAP) to assure adequate net positive suction head (NPSH) to the ECCS and containment heat removal pumps was given considerable attention recently by the NRC. The NRC staff acknowledges the licensee's claim in Section 2.6.5.1 that adequate NPSH margin is maintained for the low-pressure injection (LPI) and building spray (BS) pumps. However, based on new guidance on NPSH margin applicable to EPU reviews, including CR-3, the NRC staff needs to determine whether use of CAP could become necessary for plants requesting EPU, with or without uncertainties included in the calculations. Also, the maximum erosion zone (defined in the guidance document) needs to be addressed. The following are some recent documents from the NRC that led to the application of new guidance to EPU applications.

- Letter from NRC to Pressurized-Water Reactor Owners' Group [PWROG], "The Use of Containment Accident Pressure in Demonstrating Acceptable Operation of Emergency Core Cooling and Containment Heat Removal Pumps during Postulated Accidents," dated March 24, 2010 (ADAMS Accession No. ML100740579).

- NRC Commission Paper, SECY-11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," dated January 31, 2011 (ADAMS Accession No. ML102780586).

- NRC Staff Requirements Memorandum, "Staff Requirements – SECY-11-0014 – Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," dated March 15, 2011 (ADAMS Accession No. ML110740254).

In order to make an informed decision as to whether the licensee is in effect utilizing or not utilizing CAP when the new guidance is applied to NPSH evaluations and determine if the evaluations are in accordance with the guidance, the NRC staff requires additional information.

SCVB-3.1 Provide the basis for the NPSH required (NPSHR) of the high-pressure injection, LPI and BS pumps (tested value, extrapolation to flows other than tested flows), including flow rates assumed, and a comparison with the flow rate for the LOCA peak cladding temperature analyses. What head drop value is used for NPSHR (3% head drop or other)?

SCVB-3.2 Provide details of the method of calculating NPSH available (NPSHA) for all the above pumps (e.g., Refueling Water Storage Tank (RWST) level, containment atmospheric pressure, vapor pressure, head loss through suction piping, sump water temperature).

SCVB-3.3 Provide the results in a tabular form for both the injection phase and recirculation phase. As a minimum, include the flow rates, static head at minimum levels (RWST or sump), head loss through suction piping, containment atmosphere pressure, vapor pressure, water temperature, NPSHA, NPSHR, NPSH margin and friction losses.

SCVB-3.4 Please demonstrate that NPSH margin still exists after including the uncertainties in the required NPSH. The NRC staff, in consultation with a pump expert, determined that a 21-percent margin on the "3%-required NPSH" would envelope the uncertainties in the draft guidance document. It is acceptable to the NRC staff, if the EPU applicants desire, to use this value in lieu of performing detailed plant specific uncertainty evaluations.

SCVB-3.5 Provide a discussion of how the post-accident debris generation at CR-3 is impacted by the EPU and the resultant impact on the sump strainer head loss and on the pump NPSH evaluations.

March 2, 2012

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ADAMS Accession No: ML120550561 * By memo

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