Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.4

August 25, 2009
3F0809-02

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – Response to Individual Plant Actions Regarding the Pressurized-Water Reactor Owners Group Topical Report, BAW-2374, Revision 2, "Risk-informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping" (TAC NO. ME1799)

Reference: NRC to Crystal River Unit 3 letter dated July 31, 2009, "Crystal River Unit 3 Nuclear Generating Plant – Individual Plant Actions Regarding the Pressurized-Water Reactor Owners Group Topical Report, BAW-2374, Revision 2, "Risk-informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping (TAC NO. ME1799)""

Dear Sir:

Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc. (PEF), is submitting this correspondence to address actions resulting from the June 25, 2009, public meeting between the Nuclear Regulatory Commission (NRC) and the Pressurized-Water Reactor Owners Group (PWROG). Attachment 1 contains the response to Actions 1 through 6. Attachment 2 contains a regulatory commitment contained in this submittal.

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Supervisor, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

Jon A. Franke
Vice President
Crystal River Unit 3

JAF/dwh

Attachments: 1. Response to Actions
2. Regulatory Commitment

xc: NRR Project Manager
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PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER - UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT 1

RESPONSE TO ACTIONS

RESPONSE TO INDIVIDUAL PLANT ACTIONS REGARDING PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT BAW-2374, REVISION 2

The U.S. Nuclear Regulatory Commission (NRC) requested that within 30 days from the date of the referenced letter, each Babcock and Wilcox (B&W) licensee submit a letter providing plans to address the following items resulting from the June 25, 2009, public meeting regarding once-through steam generator (SG) tube loads under conditions resulting from postulated breaks in reactor coolant system upper hot leg large-bore piping:

1) Confirmation that its justification for continued operation for addressing tube integrity following a large break loss-of-coolant accident (LBLOCA) remains valid. (All B&W licensees)

Response: The open Corrective Action Program Action Request from 2000, and the engineering evaluation that contains the justification for continued operation associated with Once-Through Steam Generator (OTSG) tube integrity and leakage following a LBLOCA, have been reviewed for Crystal River Unit 3 (CR-3). The existing plant documentation and evaluations continue to adequately address the current condition of the OTSG tubing. Additionally, a regulatory commitment contained in License Amendment #198, "Crystal River Unit 3 – Issuance of Amendment Regarding Reroll Repair for Once-Through Steam Generator Tubing (TAC NO. MB1519)," dated September 10, 2001, requires CR-3 to perform a best estimate leakage calculation during each OTSG inspection in order to address the postulated leakage associated with circumferentially-oriented degradation under LBLOCA load conditions. The justification for continued operation remains valid.

2) Confirmation that compensatory measures, such as changes to emergency operating procedures, have been incorporated into plant procedures and operator training has been performed. (All B&W licensees)

Response: Compensatory measures completed include revision to Emergency Operating Procedures (EOPs) related to the potential loss of Reactor Coolant System (RCS) inventory from a LBLOCA. These actions included the impact from an assumed failure of a Main Steam Isolation Valve (MSIV) to isolate.

As part of the CR-3 submittal for License Amendment #198, CR-3 stated that plant-specific EOPs were consistent with the descriptions in BAW-2374 with regard to key operator actions required to transfer the Emergency Core Cooling System (ECCS) suction from the Borated Water Storage Tank (BWST) to the Reactor Building (RB) Sump. CR-3 also provided confirmation of secondary system isolation to contain primary-to-secondary leakage. Based on discussions with the B&W Owners Group and the NRC in 2006, CR-3 again revised EOPs to address this potential LBLOCA scenario to include the failure of an MSIV to isolate. The current EOPs contain guidance to isolate potential leak paths downstream of a MSIV. Training of Operations personnel on the revised EOPs was completed during their normal training cycle.

CR-3 revised the calculation for the RB flood level based on water bypassing the RB through the OTSG and into the Main Steam lines using the assumed leak rate from BAW-2374, Revision 2. The same leak rate was also used to verify that net positive suction head (NPSH) remained adequate for the Building Spray and Decay Heat pumps.

Based on the above information, the CR-3 compensatory measures related to the BAW-2374, Revision 2 LBLOCA scenario have been established and are complete.

3) Confirmation that 10CFR50.46(a)(3) reporting requirements have been satisfied. (All B&W licensees)

Response: A variety of RCS break locations have been investigated to confirm that the double-ended guillotine break in the cold leg pump discharge (CLPD) piping is the worst break for generic applications. Since postulated breaks in RCS upper hot leg large-bore piping have been determined to not be the limiting location, they are not reportable under 10CFR50.46(a)(3). Hot leg breaks are not limiting because core flows remain significantly positive during blowdown, leading to a high heat transfer compared to the CLPD breaks.

4) Confirmation that all LBLOCAs (including those in the candy-cane region of the reactor coolant system (RCS)) are considered as design basis accidents in the assessments of SG tube integrity following each SG tube inspection. (All B&W licensees)

Response: All LBLOCAs, including those in the candy-cane region of the RCS, will be considered in the CR-3 assessments of OTSG tube integrity following each future OTSG tube inspection. The site procedure for performing tube inspections and assessments has been revised to reflect that LBLOCA conditions must be evaluated during each assessment of tube integrity. The procedure revision is complete and will be issued prior to startup from the Fall 2009 Steam Generator Replacement (SGR) outage and will be in place for all future assessments. This is consistent with the implementation date for License Amendment #234, "Crystal River Unit 3 – Issuance of Amendment Regarding the Revision of the Steam Generator Portion of the Technical Specifications to Reflect the Replacement of the Steam Generators (TAC No. MD9547)."

5) Provide confirmation that the design of the replacement SGs is sufficient to withstand the loads associated with a LBLOCA including the thermal loads associated with a LBLOCA in the candy-cane region of the RCS. (Oconee and Crystal River, only)

Response: The CR-3 replacement steam generators (SGs) are designed to withstand the loads associated with a spectrum of LBLOCAs, including the thermal loads resulting from a double-ended guillotine break in the candy cane region of the hot leg. This has been demonstrated by ASME Section III Code analysis for Level D conditions for the LBLOCA.

6) Commitment to provide the structural limit associated with the most limiting LBLOCA for the replacement SGs as part of the next SG tube inspection report (required by the technical specifications) following completion of the next inspection of the tubes in the replacement SGs, unless previously submitted. (All B&W licensees)

Response: CR-3 will provide the structural limit associated with the most limiting LBLOCA for the replacement SGs as part of the next OTSG tube inspection report (required by the technical specifications) following completion of the next inspection of the tubes in the replacement SGs. This due date corresponds to 180 days after the initial entry into Mode 4 following completion of the Steam Generator tube inspections in the Fall 2011 Refueling Outage.

PROGRESS ENERGY FLORIDA, INC.

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ATTACHMENT 2

REGULATORY COMMITMENT

Regulatory Commitment

The following table identifies the action committed to by Florida Power Corporation (FPC) in this document. Any other actions discussed in the submittal represent intended or planned actions by FPC. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing and Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

Commitment	Due Date
CR-3 will provide the structural limit associated with the most limiting LBLOCA for the replacement SGs as part of the next OTSG tube inspection report (required by the technical specifications) following completion of the next inspection of the tubes in the replacement SGs.	180 days after the initial entry into Mode 4 following completion of the Steam Generator tube inspections in the Fall 2011 Refueling Outage.