Progress Energy

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

Ref: 10 CFR 50.36

November 20, 2003
3F1103-06

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

Subject: Crystal River Unit 3 – Response to Request for Additional Information Regarding License Amendment Request #280, Revision 0, Revised Improved Technical Specification 3.7.9, Nuclear Services Seawater System

Reference: PEF to NRC letter, 3F0703-04, dated July 14, 2003, Crystal River Unit 3 – License Amendment Request #280, Revision 0, Revised Improved Technical Specification (ITS) 3.7.9, Nuclear Services Seawater System

Dear Sir:

The purpose of this submittal is to provide the NRC with the Progress Energy Florida Inc. (PEF) responses to the NRC questions regarding the subject License Amendment Request (LAR) #280. The NRC questions (See Attachment) were discussed during a telephone conference held on October 9, 2003.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/lvc

Attachment:
Response to NRC Questions Regarding License Amendment Request (LAR) #280

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

A001

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 20th day of November, 2003, by Dale E. Young.

Janet Schroeder

Signature of Notary Public
State of Florida

JANET SCHROEDER
MY COMMISSION # DD 128063
EXPIRES: June 20, 2006
Bonded Thru Notary Public Underwriters

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT

Response to NRC Questions
Regarding License Amendment Request (LAR) #280

Regarding the one-time extension of the Nuclear Services Seawater System train for Crystal River Unit 3 (CR-3) discuss the following:

Question

1. In Attachment A, Page 5 of 9 (Progress Energy Florida, Inc. (PEF) to NRC letter dated July 14, 2003), it is stated that, "Informal calculations performed show that below a UHS (Ultimate Heat Sink) temperature of approximately 90°F, RWP-1 (Normal duty Nuclear Services Seawater System Pump) can provide enough flow to remove heat loads in accident conditions." Please address the following questions:

   a) Was credit taken for use of RWP-1 in the PRA (Probabilistic Risk Assessment) calculations presented in Attachment E of PEF's request? If so, please provide the Fussell-Vesely importance measure and the risk achievement worth of RWP-1 for both the baseline configuration and the plant configuration during refurbishment of RWP-2A/2B (Nuclear Services Emergency Seawater System pumps).

Response

RWP-1 is credited in the PRA for non-Engineered Safeguards (ES) Loss-of-Coolant Accident (LOCA) scenarios. The loads for these scenarios are similar to the normal plant loads for which RWP-1 is the operating pump. RWP-2A/2B are sized to handle maximum LOCA heat loads, including Reactor Building cooling, using the design basis UHS of 95°F. The "informal calculations" showed that RWP-1 could handle these loads if UHS < 90°F, but this has not been credited in the PRA models.

Question

1. b) Based on PEF's cover letter, RWP-2A/2B refurbishment seems likely during summertime. Please provide information concerning the likelihood that UHS temperature would exceed 90°F during RWP-2A/2B refurbishment. Was this situation addressed in the PRA calculations?

   Regulatory Basis: Regulatory Guide 1.177, Section 2.3.6.

Response

A review of the Gulf temperatures taken at the intake for the last 5+ years show that the temperature has slightly exceeded 90°F several times. The longest duration noted was about 24 hours. Gulf temperature is not addressed in the PRA models as it is assumed that we would shut down if temperatures exceeded design basis without supporting justification for continued operation. As stated in the response to Question 1.a above, the PRA does not credit using RWP-1 for LOCA scenarios.

Question

2. Please describe how RWP-2A/2B will be isolated during its refurbishment. Has the possibility of isolation failure (e.g., valve failure, human error during refurbishment activities) leading to a flow diversion pathway and consequential loss of all Nuclear Services Seawater been considered?

   Regulatory Basis: Regulatory Guide 1.174, Section 2.1 and Regulatory Guide 1.177, Section B, Element 1: Define the Proposed Change.

Response

The PRA includes a potential for flow diversion if a check valve fails on one of the standby pumps during normal operation. During the pump maintenance, the pipe will be isolated with a single manually operated butterfly valve. The likelihood of a diversion failure is assumed to be the same as normal operation. No special treatment was applied based on increasing the allowed outage time (AOT) from 3 to 10 days.

Internal flooding due to maintenance activities is also included in the model (generally related to heat exchanger cleaning). Typically one heat exchanger (out of four) is cleaned each week. During maintenance, the heat exchangers are isolated from the RW header using a manual valve. The flooding frequency during the RWP-2A/2B maintenance is assumed bounded by the current model.

Question

3. Attachment D, Calculation No. P-03-0001, Page 3 (PEF to NRC letter dated July 14, 2003), indicates that there is an increased likelihood of loss of NSCCC (Nuclear Services Closed Cycle Cooling Water System) cooling during refurbishment activities, and that the frequency of loss of NSCCC cooling was accordingly increased by a factor of 10. Please discuss how the frequency of loss of NSCCC cooling was estimated in the baseline PRA.

   Regulatory Basis: Regulatory Guide 1.177, Section 2.3.2.3.

Response

The loss of NSCCC (SW) frequency was calculated using a separate fault tree (including the RW-SW dependency) and applied to the baseline PRA model as a point estimate. A sensitivity study was performed to the SW tree with one RW pump out of service that showed an increase of about a factor of 3. A factor of 10 was applied to account for uncertainties. Note that the RW-SW system referenced in the submittal is the cooling source for SW and has a direct input to the loss of SW.

Question

4. Attachment A, Page 8 (PEF to NRC letter dated July 14, 2003), indicates that substantial changes for example, revision of accident sequence logic for SGTR (Steam Generator Tube Rupture) and ATWS (Anticipated Transients Without Scram), addition of a new initiating event for loss of all raw water pumps, etc., were made to the PSA (Probabilistic Safety Assessment) model as a result of the peer review completed in September 2001. ASME (American Society of Mechanical Engineers) RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Section 2 classifies such changes as "PRA upgrade." Section 5.4 of the same standard indicates that PRA upgrades shall satisfy the peer review requirements specified in Section 6 of the standard, but limited to aspects of the PRA that have been upgraded. Please, describe how ASME RA-S-2002 is being used to help assure the quality of the CR-3 PSA model.

   Regulatory Basis: Regulatory Guide 1.174, Sections 2.2.3 and 2.5, and Regulatory Guide 1.177, Section 2.3.1.

Response

As indicated in the question above, enhancements were made to the CR-3 PSA as a result of the peer certification review. However, no formal review of the CR-3 PSA has been completed using the ASME PSA Standard, and the CR-3 PSA has not had additional industry peer reviews since the NEI/B&WOG Certification Review.

CR-3 is not classifying the "updates" and enhancements made as "upgrades" since these were not methodology changes but changes to address questions raised internally or by the NEI peer review. Therefore, these "updates" and enhancements would not require a peer review.