



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

Ref: 10 CFR 50.90

May 6, 2005
3F0505-07

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

Subject: Crystal River Unit 3 – Supplemental Information and Additional Commitment Regarding License Amendment Request #289, Revision 0, Revised Improved Technical Specifications (ITS) 3.5.2, Emergency Core Cooling Systems (ECCS) - Operating, 3.6.6, Reactor Building Spray and Containment Cooling Systems, 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System and 3.7.10, Decay Heat Seawater System (TAC No. MC5631)

- References:**
1. PEF to NRC letter dated January 13, 2005, Crystal River Unit 3 - License Amendment Request #289, Revision 0, Revised Improved Technical Specifications (ITS) 3.5.2, Emergency Core Cooling Systems (ECCS) - Operating, 3.6.6, Reactor Building Spray and Containment Cooling Systems, 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System and 3.7.10, Decay Heat Seawater System
 2. PEF to NRC letter dated February 11, 2005, Crystal River Unit 3 – Supplemental Information Regarding Risk Significant Fire Zones and Fire Zone Specific Compensatory Actions for License Amendment Request #289, Revision 0, Revised Improved Technical Specifications (ITS) 3.5.2, Emergency Core Cooling Systems (ECCS) - Operating, 3.6.6, Reactor Building Spray and Containment Cooling Systems, 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System and 3.7.10, Decay Heat Seawater System

Dear Sir:

Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF), hereby submits supplemental information and an additional Regulatory Commitment for License Amendment Request (LAR) #289, Revision 0.

The supplemental information is included in Attachment A. This information includes NRC questions and Crystal River Unit 3 (CR-3) responses discussed with the NRC staff during a telecom held on April 18, 2005.

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

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A list of regulatory commitments is included in Attachment B. References 1 and 2 provided regulatory commitments to be implemented during the proposed one-time Allowed Outage Time (AOT). CR-3 will also implement the provisions described in the additional commitment contained in Attachment B during the proposed one-time AOT.

Reference 1 did not specify an approval date for LAR #289, Revision 0. CR-3 is hereby requesting an approval date of August 1, 2005 in order to determine whether the refurbishment of RWP-3B will be conducted online or during Refuel Outage 14 (14R).

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
Vice President
Crystal River Nuclear Plant

DEY/lvc

Attachments:

- A. Supplemental Information Regarding LAR #289, Revision 0
- B. List of Regulatory Commitments

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

STATE OF FLORIDA
COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF); 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.

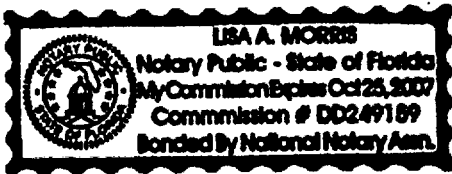
Dale E Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 6th day of May, 2005, by Dale E. Young.

Lisa A Morris

Signature of Notary Public
State of Florida



LISA A MORRIS

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

Personally Known X -OR- Produced Identification _____

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #289, REVISION 0

Supplemental Information Regarding LAR #289, Revision 0

RAI Question 1

In both Attachment A (Page 12, first paragraph) and Attachment E (Calculation No. P-05-0001, Table 1, Note 2) it is stated that the Probabilistic Risk Assessment (PRA) evaluation of the proposed license amendment assumed no maintenance will be scheduled on risk-sensitive equipment beyond that required for the refurbishment of Decay Heat Seawater System Pump RWP-3B (Nuclear Services and Decay Heat Seawater System, Decay Heat System, Decay Heat Closed Cycle Cooling Water System, Nuclear Services Closed Cycle Cooling Water, Emergency Diesel Generators, Emergency Feedwater System, Emergency Feedwater Initiation and Control System, and the Auxiliary Feedwater Pump). However, there is no regulatory commitment in Attachment F to forbid or limit maintenance on risk-sensitive equipment during the refurbishment of RWP-3B. Please resolve this apparent contradiction.

Response

Compensatory Action 1 of Attachment A, Reference 1 requires the performance of Compliance Procedure CP-253, "Power Operation Risk Assessment and Management," which will estipulate that no maintenance will be performed on risk sensitive equipment. However, Crystal River Unit 3 is making the following commitment in addition to those previously provided in References 1 and 2: Equipment and systems (including support equipment) will be designated administratively "as protected" (no planned maintenance beyond that required for the RWP-3B refurbishment activity): Nuclear Services and Decay Heat Seawater System, Decay Heat System, Decay Heat Closed Cycle Cooling Water System, Nuclear Services Closed Cycle Cooling Water, Emergency Diesel Generators, Emergency Feedwater System, Emergency Feedwater Initiation and Controls System (EFIC) and Auxiliary Feedwater Pump.

RAI Question 2

On Page 12 of Attachment A, it is stated that "...the bounding risk due to internal events for this activity [the proposed license amendment] is estimated with a Change in Core Damage Frequency (Δ CDF) of $1.5E-6$ and a sensitivity based on fire risk add about $2.72E-6$." However, Attachment E (Calculation No. P-05-0001, Table 1) indicates that the value " $1.5E-6$ " is the incremental conditional core-damage probability (ICCDP) for internal events. Attachment E (Calculation No. P-05-0001, Section 5.2) also indicates that the value " $2.72E-6$ " is an estimate of the ICCDP due to internal fires. Please, explain what the values " $1.5E-6$ " and " $2.72E-6$ " signifies (Δ CDF or ICCDP).

Comment: If these values are ICCDPs, then it is not appropriate to compare them to the risk acceptance guidelines in RG 1.174 as was done on Page 12 of Attachment A; rather, they should be compared to the risk acceptance guidelines in RG 1.177.

Response

The values were calculated by multiplying the change in instantaneous risk (Δ CDF_i) by the duration of the activity, so the value is an ICCDP for the activity. Since the activity involves a one-time activity, the delta CDF for that year is equal to the ICCDP for the activity in which this activity occurs (Δ CDF * 1yr = CDP). Reporting Δ CDF for a one-time Allowed Outage Time (AOT) extension in this manner is conservative. If the maintenance unavailability time for this

one-time activity is applied to the average maintenance unavailability of the current baseline model, the average unavailability probability for the RWP 3A/B pumps increases from .0086 to .016. Re-quantifying the model with this value increases the CDF to 8.16E-06/yr resulting in a Δ CDF of 6.7E-07/yr. This value is very low and supports the acceptability of the AOT request per RG 1.174.

RAI Question 3

On Page 12 of Attachment A, only one type risk metric was provided (apparently the internal events and internal fires ICCDP values associated with the proposed license amendment). Attachment E (Calculation No. P-05-0001, Table 1) also provides the incremental conditional large early release probability (ICLERP) for internal events. Please provide either the ICLERP associated with internal fires or a qualitative assessment of the proposed license amendment's impact on the likelihood of large early release following internal fires. In addition, in accordance with RG 1.177, Section 2.4, "Acceptance Guidelines for Technical Specifications (TS) Changes," please provide the risk metrics discussed in RG 1.174 (total Δ CDF versus total CDF, and total Large Early Relief Frequency (LERF) versus total LERF) associated with the proposed license amendment.

Response

CR-3 does not have a fire PRA model that can be used to quantify the effect of the postulated fire scenario on LERF. However, because the predominant contributor to LERF for CR-3 are scenarios based on steam generator tube ruptures (SGTR) or interfacing system LOCAs (ISLOCA), the LERF impact is estimated to be very low. A fire in the "A" 4160V switchgear room would not contribute to an increase in the frequency for either of these initiating events, and the likelihood of a fire occurring in the switchgear room coincident with a SGTR or ISLOCA is also very small. Therefore, any increase in LERF will be very small. Given the very low change in LERF due to the internal events assessment, additional contributions from the fire are expected to support the acceptability of the AOT request per the regulatory guide.

RAI Question 4.1

Attachment A, Page 12 and Attachment F indicate that compliance procedure CP-253, "Power Operation Risk Assessment and Management" will be performed. Either provide CP-253 or summarize its contents, indicating how it satisfies the guidance in RG 1.177, Section 2.3, "Tier 3: Risk-Informed Configuration Risk Management" and RG 1.177, Section 2.3.7, "Contemporaneous Configuration Control." The following questions should be addressed:

Is CP-253 intended to satisfy the requirements of 10 CFR 50.65, i.e., is CP-253 the mechanism for complying with Paragraph (a)(4) of the Maintenance Rule?

Response

Yes, CP-253 provides the guidance for performing configuration risk management on CR-3 risk assessments.

RAI Question 4.2

Does CP-253 relate to Chapter 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants?" If so, describe any differences between CP-253 and NUMARC 93-01, Chapter 11.

Response

Yes, CP-253 is intended to implement the guidance provided by NUMARC 93-01, Chapter 11. The risk evaluation is a blended approach using the plant PRA for quantitative insights and qualitative input for activities and conditions outside of the scope of the PRA or not impacting CDF. Thresholds based on CDF/ICDP are used as the quantitative tools for directing risk management actions while at power.

RAI Question 4.3

In CP-253, what provisions are provided for assessing the need for additional actions after the discovery of additional equipment-out-of-service conditions while RWP-3B is being refurbished? What criteria or guidelines are provided to help decide whether or not additional actions are either needed or not needed? Who makes the decision (the operating crew, plant management, utility management)? Does CP-253 impose any timeliness limits on reaching the decision about additional actions?

Response

Risk sensitive equipment listed in the response to Question 1 will be kept available during the RWP-3B refurbishment. Equipment not listed will be evaluated using the guidance provided in CP-253. Any activities which challenge the conclusions of the RWP-3B submittal will be re-scheduled. Emergent items would also be evaluated and appropriate actions taken based on the risk insights. Risk management actions are generally recommended to Operations by Work Controls and PRA. Many risk management actions are already built into the plant procedures. If an "ORANGE" risk threshold is anticipated to be exceeded, Plant General Manager approval is required. Risk management actions are expected to be implemented prior to the work activity. In the case of emergent problems, additional actions would be implemented as soon as possible based on the conditions.

RAI Question 4.4

What tools or techniques are used to help assess the risk of various plant configurations, e.g., real-time risk monitoring software. What quality assurance activities have been taken to ensure that these tools and techniques are adequate, e.g., periodic updates, reviews, etc.

Response

For the quantitative evaluation of risk, CR-3 uses the Equipment-Out-Of-Service (EOOS) software provided by the Electric Power Research Institute (EPRI). The software has been implemented in accordance with Progress Energy Software Quantity Assurance Procedures, and the installation is controlled by the Information Technology (IT) department. The PRA models used by EOOS are

equivalent to the models used in development of this submittal and meet the same quality requirements. They are implemented using the Progress Energy Engineering Change procedure and a calculation which includes establishment of the baseline risks, thresholds, and benchmarks. The baseline files are maintained on a read only directory.

REFERENCES

1. PEF to NRC letter dated January 13, 2005, Crystal River Unit 3 - License Amendment Request #289, Revision 0, Revised Improved Technical Specifications (ITS) 3.5.2, Emergency Core Cooling Systems (ECCS) - Operating, 3.6.6, Reactor Building Spray and Containment Cooling Systems, 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System and 3.7.10, Decay Heat Seawater System
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PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #289, REVISION 0

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Progress Energy Florida (PEF) in this document. Any other actions discussed in the submittal represent intended or planned actions by PEF. 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 |
|---|---|
| Crystal River Unit 3 is making the following commitment in addition to those previously provided in References 1 and 2: Equipment and systems (including support equipment) will be designated administratively "as protected" (no planned maintenance beyond that required for the RWP-3B refurbishment activity): Nuclear Services and Decay Heat Seawater System, Decay Heat System, Decay Heat Closed Cycle Cooling Water System, Nuclear Services Closed Cycle Cooling Water, Emergency Diesel Generators, Emergency Feedwater System, Emergency Feedwater Initiation and Controls System (EFIC) and Auxiliary Feedwater Pump. | During one-time extended (greater than 72 hours) RWP-3B maintenance |