

October 21, 2004

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING A
ONE-TIME EXTENSION OF AN EMERGENCY FEEDWATER SYSTEM
TRAIN COMPLETION TIME (TAC NO. MC1586)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 214 to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). The amendment consists of changes to the existing Technical Specifications (TS) in response to your letter dated December 9, 2003, as supplemented by letter dated September 16, 2004. The proposed change would revise TS 3.7.5 to allow an emergency feedwater train to be inoperable for up to 14 days, on a one-time basis, to allow performance of emergency feedwater pump (EFP-3) repairs. This 14-day proposed one-time completion time would expire on March 31, 2005.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Brenda L. Mozafari, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:

1. Amendment No. 214 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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Crystal River Nuclear Plant, Unit 3

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CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION,
CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Florida Power Corporation, et al. (the licensees), dated December 9, 2003, as supplemented by letter dated September 16, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 214, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/KJabbour for/

Michael L. Marshall, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 21, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 214

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following page of the Appendix "A" Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains vertical lines indicating the areas of change.

Remove

3.7-9

Insert

3.7-9

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-72
FLORIDA POWER CORPORATION, ET AL.
CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT
DOCKET NO. 50-302

1.0 INTRODUCTION

By application dated December 9, 2003, as supplemented by letter dated September 16, 2004, Florida Power Corporation (the licensee, also doing business as Progress Energy Florida, Inc.) proposed changes to the Crystal River Unit 3 (CR-3) Technical Specifications (TS) in accordance with 10 CFR 50.90.

The proposed change would increase, on a one-time basis, the Completion Time (CT) to restore an inoperable Emergency Feedwater (EFW) train. Specifically, the proposed change would revise the CT for TS 3.7.5, Condition B, Required Action B.1 from 72 hours from the time the affected train is declared inoperable and 10 days from the discovery of the failure to meet the limiting condition for operation (LCO) to 14 days. The CT extension may only be invoked once and remains applicable until March 31, 2005.

The reason for the licensee's request is that inservice testing of the EFW Pump (EFP-3) speed increaser shows an elevated vibration level. Since the normal duration of the repair activity would be greater than the 72-hour CT specified in TS 3.7.5, the repair work must be done in a shutdown MODE unless a one-time extension of the CT for up to 14 days is approved. Approval of the proposed change would allow the performance of the repair online, and will prevent a potential forced shutdown.

The September 16, 2004, supplemental letter provided additional information that clarified the application but did not expand the scope of the application as originally noticed and did not change the NRC staff's original proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The NRC staff finds that the licensee, in Attachment A, Page 1, of its submittal, identified the applicable regulatory requirements.

2.1 Description of System/Components and Current Requirements/Desired Change

The EFW system supplies flow to the Once-Through Steam Generators (OTSGs) when main feedwater is not available. With respect to accident mitigation, the EFW system is relied upon for small-break loss-of-coolant accidents, loss of main feedwater, loss of offsite power, main feedwater breaks, main steamline breaks, and anticipated transients without scram (ATWS). The EFW system has one 100-percent capacity turbine-driven pump (EFP-2) and one 100-percent capacity diesel-driven pump (EFP-3). These pumps are automatically started and controlled by the Emergency Feedwater Initiation and Control (EFIC) system. In addition, a

safety-grade 100-percent capacity motor-driven pump (EFP-1) is available for manual initiation; this pump is normally powered from offsite power and may be manually loaded on the "A" train emergency diesel generator (EDG) if electrical loading capacity is available.

As additional defense-in-depth, the Auxiliary Feedwater (AFW) System motor-driven pump (FWP-7) can be manually initiated if required. This pump is powered by either offsite power or a non-safety-related diesel generator (MTDG-1).

Recently, EFP-3 has been exhibiting an elevated vibration on the speed increaser located between the diesel engine and the pump. Although the pump is currently operable, the licensee desires to perform corrective maintenance, in the form of realignment, to reduce the vibration prior to the next refueling outage, which is presently scheduled for Fall 2005. The licensee states that the preferred time to perform this maintenance would be the fourth quarter of 2004. The NRC staff agrees that the sooner it is performed, the better.

TS 3.7.5, "Emergency Feedwater System," requires that two EFW trains be operable. If one train is inoperable, then it must be restored to operable status within 72 hours from the time it was declared inoperable and within 10 days of discovery of failure to meet the LCO. The second 10-day limitation is designed to prevent indefinite continued operation while not meeting the LCO. The licensee's estimated time to perform the proposed maintenance activity is 7 days. The licensee, following Institute of Nuclear Power Operations (INPO) recommended practice, would like to obtain a CT (one-time only in this submittal request) of twice the estimated time for completion of the proposed corrective maintenance procedure. Therefore, a one-time relaxation of the TS 3.7.5 CT to 14 days is being requested by the licensee in order to allow performance of the proposed realignment online.

2.2 Description of the Proposed License Amendment Request

The proposed change adds a note to the 72-hour and 10-day CTs of TS 3.7.5, Condition B, Required Action B.1, as follows:

*On a one-time basis, an EFW train may be inoperable for up to 14 days to allow performance of EFW Pump (EFP-3) repairs. The ability to apply the 14-day Completion Time will expire on March 31, 2005.

2.3 Applicable Regulatory Criteria/Guidelines

The regulatory criteria/guidelines on which the NRC staff can base its acceptance are:

- Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," describes a risk-informed approach, acceptable to the NRC staff, for assessing the nature and impact of proposed licensing basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.
- RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," describes an acceptable risk-informed approach specifically for

assessing proposed TS changes in allowed outage times (AOTs). Note that the phrase “Completion Time” used in the licensee’s TS is equivalent to the phrase “allowed outage time” used in RG 1.177. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

One acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary TS changes, is to show that the proposed changes meet five key principles stated in RG 1.174, Section 2 and RG 1.177, Section B:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission’s Safety Goal Policy Statement.
5. The impact of the proposed change should be monitored using performance measurement strategies.

For permanent TS changes, RG 1.174 and RG 1.177 provide numerical risk acceptance guidelines that are helpful in determining whether or not the fourth key principle has been satisfied. These guidelines are not to be applied in an overly prescriptive manner; rather, they provide an indication, in numerical terms, of what is considered acceptable. The intent in comparing risk results with the risk acceptance guidelines is to demonstrate with reasonable assurance that the fourth key principle has been satisfied.

For temporary TS changes, examination of the risk metrics identified in RG 1.174 and RG 1.177 provides insight about the potential risk impacts, even though neither of these RGs provides numerical risk acceptance guidelines for evaluating temporary TS changes against the fourth key principle. The NRC Staff Requirements Memorandum (SRM) issued March 19, 1998, in response to SECY-97-287 provides some additional guidance concerning the use of risk acceptance guidelines to evaluate temporary changes to plant configurations:

At present, except for the special case of Technical Specification changes, only time-averaged guidelines will be used. In assessing these acceptance guidelines for temporary plant configurations staff should weigh the merits of temporary changes that may lead to improved safety, and should ensure that appropriate compensatory measures are taken into account that could mitigate the conditional CDF [core damage frequency] and LERF [large early release frequency].

Based on this SRM, coupled with RG 1.174 and RG 1.177, the NRC staff concludes that a proposed temporary TS change meets the fourth key principle of risk-informed decisionmaking if its associated risk metrics:

- Satisfy the risk acceptance guidelines in RG 1.174 and RG 1.177, or
- Are not substantially above the risk acceptance guidelines in RG 1.174 and RG 1.177, and effective compensatory measures to lower risk are implemented while the temporary TS change is in effect.

3.0 TECHNICAL EVALUATION

The NRC staff's comparison of the licensee's proposed license amendment for a one-time extension of the EFW System train CT against the five key principles of risk-informed decisionmaking is presented in the following sections.

3.1 Traditional Engineering Evaluation

The traditional engineering evaluation presented below addresses the first three key principles of the NRC staff's philosophy of risk-informed decisionmaking, which concern compliance with current regulations, evaluation of defense-in-depth, and evaluation of safety margins.

3.1.1 Compliance with Current Regulations

The licensee does not propose to deviate from existing regulatory requirements, and compliance with existing regulations is maintained by the proposed one-time change to the TS requirements.

3.1.2 Evaluation of Defense-in-Depth

In the recent licensee performance evaluation of EFP-3, it was concluded by the licensee that the pump is fully capable of supporting plant operation. Even though EFP-3 has exhibited elevated vibration levels, the licensee found it to be fully capable of performing its safety function. EFP-3 is not presently on an increased testing frequency since its present operational status is consistent with the requirements of the licensee's Operating and Maintenance (OM) Code, Part 6, and the CR-3 Inservice Testing Program.

The licensee states that during the requested relaxed 14-day CT the redundant "B" train pump, EFP-2, will be available to provide OTSG cooling during emergency conditions. EFP-1 is safety grade and, although it is not automatically initiated, it will also be available and capable of OTSG cooling for all design-basis events where offsite power remains available or where sufficient EDG capacity is available. FWP-7 is also capable of providing OTSG cooling for all but the most limiting design-basis events if offsite power is available or its non-safety-grade diesel generator is available. The licensee's Emergency Operating Procedures incorporate the use of EFP-1 and FWP-7 if EFP-2 and EFP-3 are not available. The NRC staff judges that this degree of defense-in-depth is adequate.

To ensure, however, that the CR-3 defense-in-depth capabilities and the assumptions in the risk assessment are maintained during the proposed one-time relaxed CT, the licensee states that it will continue the performance of 10 CFR 50.65(a)(4) assessments before performing maintenance or surveillance activities. Other compensatory actions include: use of pre-job

briefings and periodic operator walkdowns to assess the status of risk-sensitive equipment in the redundant train, use of operator walkdowns to assess and limit transient combustibles in risk-significant fire areas, and performance of no elective maintenance in the switchyard that would challenge the availability of offsite power to the engineered safeguards (ES) buses. The NRC staff judges that these compensatory actions are adequate.

The licensee states that the proposed deletion of the 10-day CT is consistent with TS Task Force (TSTF)-439, Revision 1, which deletes the second CT from several LCOs and corresponding Bases, including the TS 3.7.5, Required Action B.1, 10-day CT. TSTF-439 discusses the principle that the acceptability of deleting the 10-day CT is based on the inclusion of structures, systems, and components (SSCs) in the scope of the Maintenance Rule, which would determine if continuous multiple entries into the ACTIONS of the TS result in unacceptable unavailability of these SSCs. The NRC staff agrees that the Maintenance Rule provides an adequate safety level.

3.1.3 Evaluation of Safety Margins

The licensee has not requested an exemption from any code or standard in conjunction with the proposed CT change. As a result, the NRC staff concludes that the proposed CT change meets applicable codes and standards. In addition, the licensee stated that the EFW system is not an initiator of design-basis accidents, and that the proposed CT change does not result in changes to the design, physical configuration of the plant, or the assumptions made in the plant safety analysis. Therefore, the NRC staff finds that the proposed TS change maintains adequate safety margins because the acceptance guidelines in Standard Review Plan (SRP) Chapter 16.1, Section II.A.2 have been satisfied.

3.2 Risk Evaluation

The risk evaluation presented in the following sections addresses the last two key principles of the NRC staff's philosophy of risk-informed decisionmaking, which deal with changes in risk and performance measurement strategies. These key principles were evaluated by using the three-tiered approach described in Chapter 16.1 of the SRP and in RG 1.177.

- Tier 1: The first tier evaluates the licensee's probabilistic risk/safety assessment (PRA/PSA) and the impact on plant operational risk, as expressed by the change in CDF and the change in LERF. The change in risk is compared against the acceptance guidelines presented in RG 1.174. The first tier also attempts to ensure that plant risk does not increase unacceptably during the period when equipment is out-of-service (OOS) per the proposed license amendment, as expressed by the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP).
- Tier 2: The second tier addresses the need to preclude potentially high-risk plant configurations that could result if equipment, in addition to that associated with the proposed license amendment, are taken OOS simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The objective of this part of the review is to ensure that appropriate restrictions on

dominant risk-significant plant configurations associated with the CT relaxation are in place.

- Tier 3: The third tier addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and taking appropriate compensatory measures to avoid such configurations. The purpose of the CRMP is to ensure that equipment removed from service prior to or during the proposed relaxed CT period will be appropriately assessed from a risk perspective.

3.2.1 Tier 1: PRA Capability and Insights

The Tier 1 NRC staff review involved two aspects: (1) evaluation of the quality of the PRA and its application to the proposed CT extension, and (2) evaluation of the PRA results and insights stemming from its application.

3.2.1.1 Quality of the CR-3 PSA

The PSA used to support the licensee's submittal is a revision and extension of the original Level 1 PSA study completed in 1987 (NRC staff Safety Evaluation supporting CR-3 License Amendment No. 212, May 18, 2004), which was submitted to the NRC staff and reviewed by Argonne National Laboratory in NUREG/CR-5245. This original work, which addressed internal initiating events, was revised and augmented to include internal floods and a limited scope Level 2 PSA, and submitted in response to Generic Letter 88-20, "Individual Plant Examination (IPE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)." The licensee submitted a PSA study of external initiating events in response to Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities." The licensee's IPE was submitted to the NRC staff on March 9, 1993, and an NRC staff SE was issued on June 30, 1998. Revision 1 of the licensee's IPEEE was submitted to the NRC staff on March 24, 1997, and an NRC staff SE was issued on January 11, 2001.

Subsequent revisions to the PSA models have been performed by qualified individuals with knowledge of PSA methodology and plant systems. Involvement by engineering and operations personnel, by providing input and reviewing results, was obtained when needed based on the scope of the changes being implemented. The PSA models have been updated for various reasons, including plant changes and modifications, procedural changes, accrual of new plant data, discovery of modeling errors, and advances in PSA technology.

The CR-3 PSA model and documentation were subjected to the industry peer certification review process in September 2001. In preparation for this peer review, an external consultant was hired to develop system notebook documentation. This required a review of the system models against plant drawings and procedures and identification of any inconsistencies with the models. Items identified from this review were considered and dispositioned. The internal flooding and common cause failure analyses were updated to reflect current industry methodologies and data sources. An internal review of the PSA model elements and their corresponding documentation was conducted to assure that the model and documentation reflected the plant design properly.

The industry peer certification review was conducted by a diverse group of PSA engineers from other Babcock and Wilcox (B&W) plants, industry PSA consultants familiar with the B&W plant design, and a representative from INPO. The certification review covered all aspects of the internal events PSA model and the administrative processes used to maintain and update the model. This review generated specific recommendations for model changes to correct errors, as well as guidance for improvements to processes and methodologies used in the CR-3 PSA model, and enhancements to the documentation of the model and the administrative procedures used for model updates.

In 2004, the NRC staff assessed the quality of the CR-3 PSA used in support of a license amendment to support a one-time extension of the CT for the Nuclear Services Seawater System. The SE for License Amendment No. 212, issued May 18, 2004, indicates that the risk analysis used in support of the license amendment was of sufficient quality to support the present license amendment application.

Based on its review of the above information, the NRC staff finds that the licensee has satisfied the intent of RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Sections 2.2.3 and 2.5), and SRP Chapter 19.1, and that the quality of the CR-3 PSA is sufficient to support the risk evaluation provided by the licensee in the proposed license amendment.

3.2.1.2 Evaluation of PRA Results and Insight

The baseline CDF and LERF for CR-3 are 7.49×10^{-6} /year and 3.42×10^{-7} /year, respectively, for internal initiating events and internal floods. With a 14-day CT, the changes in CDF and LERF were computed by the licensee to be 4.70×10^{-7} /year and 5.00×10^{-9} /year, respectively. Per RG 1.174, these changes are considered to be very small.

Also for internal initiating events and internal floods, the ICCDP is calculated by the licensee to be 5.48×10^{-7} . This value is close to the RG 1.177 guideline value of 5×10^{-7} and acceptable to the NRC staff because the proposed change is a temporary change, and the licensee has proposed compensatory measures. The ICLERP is calculated by the licensee to be 6.37×10^{-9} , less than the RG 1.177 guideline value of 5×10^{-8} , and is, therefore, acceptable to the NRC staff.

The impact of the proposed CT change on external event risk was assessed by using qualitative methods as described in the following paragraphs.

Seismic Risk: CR-3 was categorized as a reduced-scope plant in NUREG-1407. The licensee used the Electric Power Research Institute's (EPRI) seismic margins assessment methodology as described in EPRI NP-6041-SL, with a Review Level Earthquake (RLE) having a 0.1g peak ground acceleration. The seismic IPEEE evaluation took credit for plant modifications and activities that had been identified under Unresolved Safety Issue (USI) A-46 program, but were not yet implemented when the IPEEE was submitted. The plant modifications and activities that were credited were subsequently implemented, and the USI A-46 program was closed in August 2000. Since the seismic margins approach was used, no quantitative seismic risk evaluation was made.

HFO Risk: The licensee evaluated high winds, floods, and other (HFO) events (hurricanes, tornados, external floods, transportation accidents, and nearby facility accidents) using the progressive screening approach described in NUREG-1407 and NUREG/CR-5042. Since CR-3 was designed prior to the issuance of the 1975 SRP, the plant was not designed according to the SRP; however, analyses were performed to determine if the plant design conforms to the 1975 SRP criteria. The licensee did not quantitatively estimate the contribution to CDF from HFO events in the IPEEE since these events were screened out on the basis of low occurrence frequency using the NRC-approved NUREG-1407 screening approach.

Fire Risk: A review of the licensee's IPEEE results indicates that internal fires are significant contributors to the overall CDF. The licensee used the Five Induced Vulnerability Evaluation (FIVE) methodology, developed by EPRI, to assess fire risks in the IPEEE. EFP-3 was installed after the IPEEE was completed. The use of FWP-7 was included in the IPEEE, but MTDG-1 and control room operation were not considered.

Since the IPEEE fire analysis has not been updated to reflect the current plant configuration, the licensee provided a qualitative defense-in-depth assessment of the potential fire risks due to the proposed change. The licensee identified several fire areas that contain cables supporting various EFW equipment and/or support equipment. By postulating fires in these areas, and considering protection, separation, detection, and suppression capabilities, fire areas were identified that could potentially impact plant capabilities with EFP-3 OOS. The licensee performed an evaluation for each fire area/zone. Six areas/zones were identified where EFP-3 would be the only pump demonstrated to be free from fire damage and addressed by fire response procedure OP-880A, Appendix "R" Post-Fire Safe Shutdown Information. For three of these areas/zones, either EFP-1 or FWP-7 were demonstrated to remain free from fire damage and would be available to provide feedwater. Because EFP-3 is the normally available EFW source, the use of these pumps is not proceduralized in OP-880A for these zones. However, if EFP-3 were unavailable, plant conditions that involve a loss of all feedwater would lead operators to emergency operating procedures (EOPs) that include provisions and direction for use of EFP-1 and FWP-7.

The following table lists the six areas/zones of highest concern with EFP-3 OOS. For each zone, the availability of the four sources of EFW and AWF during a fire was assessed. EFP-3 is normally available in all six areas/zones, but will be OOS for maintenance during the extended maintenance period. EFP-2 is unavailable for fires in all six of these areas/zones. The table does not include area/zones where the fire itself makes EFP-3 inoperable because removing EFP-3 from service does not impact EFP-3 capability for that fire. Also, the risk evaluation determined that there was little or no increase in risk when EFP-2 was available for a fire in a particular area/zone. If EFP-2 remains available for a given fire, the EFW safety function continues to be met with EFP-3 OOS.

Fire Area	Description	Train "B" EFW EFP-2	Train "A" EFW EFP-3	Train "A" EFW EFP-1	AFW FWP-7
CC-95-101	Control Complex Elevation-95	U	A	U	A*
CC-108-105	"B" Battery Charger Room	U	A	U	U*
CC-108-107	"B" ES 4160v Switchgear Room	U	A	U	U*
CC-124-113	"C" EFIC Room	U	A	A*	U*
CC-124-115	"B" EFIC Room	U	A	U	A*
CC-124-116	"B" ES 480v Switchgear Room	U	A	U	U*

- A available given fire in this zone
- A* available but not proceduralized for use during a fire; directions for use in EOPs
- U unavailable given fire in this zone
- U* unavailable per Appendix R criteria, but may be available under directions from EOPs and use of MTDG-1

For area/zone CC-124-113, the fire does not impact the EDGs. With two EDGs available, adequate capacity exists to supply EFP-1 in addition to other equipment required for safe shutdown. For zones CC-95-101 and CC-124-115, offsite power is available to FWP-7. Therefore, CR-3 has an alternate supply of feedwater for a fire in each of these areas/zones. For fires in the remaining three areas/zones (CC-108-105, CC-108-107, CC-124-116), no source of feedwater has been demonstrated to remain free from damage with EFP-3 OOS. This conclusion is based on the safe shutdown analysis done for Appendix R. The safe shutdown analysis does not take credit for MTDG-1; however, MTDG-1 may be available to power FWP-7 in some circumstances. The impact of fire on FWP-7 and MTDG-1 control and power circuitry was not analyzed for these areas/zones and, therefore, FWP-7 is not credited or addressed by OP-880A. However, FWP-7 would be used, if available, under direction of the EOPs should plant conditions require AFW.

In addition to the six areas/zones that have the highest increase in risk due to the planned EFP-3 extended maintenance, the licensee evaluated which fire areas/zones have the highest risk of core damage under any circumstance. Two of the six areas/zones previously identified above (CC-108-105 and CC-108-107) are among the highest CDF fire areas/zones. There are four additional fire areas/zones that have comparable or higher CDF values. These areas/zones are listed in the following table.

Fire Area	Description	CDF (per year)
CC-108-106	“A” Battery Charger Room	1.49 x 10 ⁻⁵
CC-108-108	“A” ES 4160v Switchgear Room	7.31 x 10 ⁻⁶
CC-124-117	“A” ES 480v Switchgear Room	3.79 x 10 ⁻⁶
CC-108-102	Hallway and Remote Shutdown Room	2.66 x 10 ⁻⁶

In order to gain further insight into the impact on fire risk due to the proposed CT change, the NRC staff performed confirmatory risk calculations. A review of the licensee’s IPEEE indicated that the CDF, due to fires in three of the six areas/zones (CC-95-101, CC-124-113, and CC-124-116), was below the FIVE methodology screening value of 10⁻⁶/year. The total CDF arising from fires in the remaining three zones was estimated at approximately 9.7 x 10⁻⁶/year, as shown below:

Fire Area	Description	Fire Frequency (per year)	CDF (per year)
CC-108-105	“B” Battery Charger Room	4.03 x 10 ⁻⁴	2.72 x 10 ⁻⁶
CC-108-107	“B” ES 4160v Switchgear Room	2.27 x 10 ⁻⁴	6.79 x 10 ⁻⁶
CC-124-116	“B” ES 480v Switchgear Room	1.90 x 10 ⁻⁴	1.76 x 10 ⁻⁷

As previously discussed, EFP-3 was installed after the IPEEE was completed. Therefore, the above results approximate the fire risks involved when EFP-3 is removed from service for maintenance. The ICCDP due to these fires is bounded by:

$$\text{ICCDP}(\text{fire}) = 9.7 \times 10^{-6}/\text{year} * (14 \text{ days}/365 \text{ days}) = 3.7 \times 10^{-7}$$

Combining the ICCDP determined by the licensee for internal initiating events and internal floods with the NRC staff-approximated ICCDP for internal fires indicates that the total ICCDP associated with the proposed change is approximately 9.2 x 10⁻⁷. This value is above the RG 1.177 guideline value of 5 x 10⁻⁷, but is acceptable to the NRC staff because the proposed change is a temporary change, and the licensee has proposed compensatory measures (refer to the Tier 2 evaluation below) that are specifically oriented towards reducing fire risks during the proposed extended preventative maintenance outage of EFP-3.

Considering the information presented above, including the conservatism and uncertainties in the analysis, the NRC staff concludes that the proposed one-time increase in the CT to allow preventative maintenance of EFP-3 results in an acceptable increase in risk that is very small and consistent with the NRC's Safety Goal Policy Statement. Therefore, the NRC staff finds that the licensee's first tier risk evaluation, as described in Chapter 16.1 of the SRP and RG 1.177, is acceptable.

3.2.2 Tier 2: Avoidance of Risk-Significant Plant Configurations

The second tier evaluates the capability of the licensee to recognize and avoid risk-significant plant configurations that could result if equipment, in addition to that associated with the proposed license amendment, is removed from service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved.

In order to avoid the emergence or persistence of risk-significant configurations during EFP-3 refurbishment activities, the licensee reviewed the PSA results to identify the following compensatory measures [Reference 2]:

- The licensee will perform procedure CP-253, "Power Operation Risk Assessment and Management," which requires both a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities. This procedure uses the Level 1 PRA model to evaluate the impact of maintenance activities on CDF. The licensee will not plan any maintenance that results in "Higher Risk" (10^{-6} Incremental Core Damage Probability) during extended EFW maintenance.
- The following equipment and systems (including support equipment) will be designated as "protected" (no planned maintenance or discretionary equipment manipulation): redundant EFW equipment, AFW (FWP-7 and MTDG-1), Control Complex Cooling, High and Low Pressure Injection, EDGs, Service Water, Raw Water, and the Appendix R chiller.
- The licensee will perform initial and daily walkdowns of the "B" train EFW system, EFP-1, the AFW System and associated power supply switchgear.
- The licensee will not schedule elective maintenance in the switchyard that would challenge the availability of offsite power.
- The licensee will not initiate an EFW extended preventative maintenance outage if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.
- Operator awareness will be increased on the need for and use of redundant EFW equipment, EFIC, AFW System, and Appendix R chiller. This will be accomplished by on-shift operating crew review of EOP-14, Enclosure 7, "EFW Pump Management," and OP-880A, Enclosure 38, "Appendix R Chiller Start."
- The following fire risk reduction actions will be taken for the areas listed below: limit hot or spark-producing work, initial walkdowns of transient combustibles, and hourly fire watch will monitor for the presence of transient combustible material.
 - CC-95-101, Control Complex Elevation-95
 - CC-108-105, "B" Battery Charger Room
 - CC-108-107, "B" ES 4160v Switchgear Room
 - CC-1124-113, "C" EFIC Room
 - CC-124-115, "B" EFIC Room

- CC-124-116, "B" ES 480v Switchgear Room
 - CC-108-106, "A" Battery Charger Room
 - CC-108-108, "A" ES 4160v Switchgear Room
 - CC-124-117, "A" ES 480v Switchgear Room
 - CC-108-102, Hallway and Remote Shutdown Room
- An hourly fire watch will be established in fire areas that are considered risk-significant:
 - CC-95-101, Control Complex Elevation-95
 - CC-108-105, "B" Battery Charger Room
 - CC-108-107, "B" ES 4160v Switchgear Room
 - CC-1124-113, "C" EFIC Room
 - CC-124-115, "B" EFIC Room
 - CC-124-116, "B" ES 480v Switchgear Room
 - CC-108-106, "A" Battery Charger Room
 - CC-108-108, "A" ES 4160v Switchgear Room
 - CC-124-117, "A" ES 480v Switchgear Room
 - CC-108-102, Hallway and Remote Shutdown Room

The review of PRA results to identify compensatory measures demonstrates the licensee's ability to recognize and avoid risk-significant plant configurations. Therefore, the NRC staff finds that the licensee's second tier risk evaluation, as described in Chapter 16.1 of the SRP and RG 1.177, is acceptable.

3.2.3 Tier 3: Risk-Informed Configuration Risk Management

The third tier assesses the licensee's program to ensure that the risk impact of OOS equipment is appropriately evaluated prior to performing any maintenance activity. The need for this third tier stems from the difficulty of identifying all possible (modeled) risk-significant configurations under the second tier that could be encountered.

To ensure that defense-in-depth capabilities and the assumptions in the PSA are maintained during the proposed one-time relaxed CT, certain compensatory measures are implemented during the EFP-3 refurbishment activity. Additionally, 10 CFR 50.65(a)(4) requires the licensee to assess and manage the increase in risk that may result from proposed maintenance activities. The NRC staff notes that the licensee maintains a real-time risk-monitoring capability.

Therefore, the NRC staff finds that the licensee's third tier risk evaluation, as described in Chapter 16.1 of the SRP and RG 1.177, is acceptable.

3.3 NRC Staff Findings

The NRC staff finds that the licensee's proposed change to revise, on a one-time basis, the CT for TS 3.7.5, Condition B, Required Action B.1, to 14 days, until March 31, 2005, is acceptable since the five key principles of risk-informed decisionmaking identified in RG 1.174 and RG 1.177 have been satisfied.

4.0 REGULATORY COMMITMENTS

During the proposed one-time 14-day EFW Train maintenance to be performed on EFP-3, the following actions are committed to by the licensee:

The licensee will perform procedure CP-253, "Power Operation Risk Assessment and Management," which requires both a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities. This procedure uses the Level 1 PSA model to evaluate the impact of maintenance activities on CDF. The licensee will not plan any maintenance activities that result in an Incremental Core Damage Probability greater than 1E-06 for this one-time maintenance activity.

The following equipment and systems (including support equipment) will be designated by the licensee administratively as "protected" (no planned maintenance or discretionary equipment manipulation): Redundant EFW Equipment, AFW (FWP-7 and MTDG-1), Control Complex Cooling, High and Low Pressure Injection, EDGs, Service Water, Raw Water, and the Appendix R Chiller.

The licensee will perform pre-job and periodic walkdowns of the "B" train EFW system, EFP-1, and the AFW.

The licensee will not schedule elective maintenance in the switchyard that would challenge the availability of offsite power.

The licensee will not initiate an EFW extended preventive maintenance outage if adverse weather, as designated by Emergency Preparedness Procedures, is anticipated.

Operator awareness will be increased on the need for and use of redundant EFW equipment, EFIC, AFW System, and Appendix R Chiller.

Fire risk reduction actions will be taken, including: limiting hot spot or spark-producing work, and pre-job and periodic walkdowns to assess and limit transient combustibles in the following key fire-important areas: Control Complex Elevation-95, "B" Battery Charger Room, "B" ES 4160v Switchgear Room, "C" Emergency Feedwater Initiation and Controls System (EFIC) Room, "B" EFIC Room, and "B" ES 480v Switchgear Room.

An hourly fire watch will be established in the above fire areas that are considered risk-significant during the planned EFP-3 maintenance.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (i.e., items requiring prior NRC approval of subsequent changes).

5.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

6.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (69 FR 16620). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 - License Amendment Request #281, Revision 0, Revised Improved Technical Specification (TS) 3.7.5, Emergency Feedwater System," 3F1203-03, ADAMS Accession No. ML033510556, December 9, 2003.
2. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, "Crystal River Unit 3 - Supplemental Information and Regulatory Commitments for License Amendment Request #281, Revision 0, Revised Improved Technical Specification (TS) 3.7.5, Emergency Feedwater System," 3F1203-03, ADAMS Accession No. ML04272043, September 16, 2004.

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