September 15, 2005

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 RE: TEMPORARY EXTENSION OF THE NUCLEAR SERVICES SEAWATER SYSTEM TRAIN COMPLETION TIME (TAC NO. MC5631)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 221 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 January 13, 2005, as supplemented by letters dated February 11, May 6, and June 9, 2005.

The amendment revises the TS to revise the Completion Time (CT) for CR-3 Improved TS 3.5.2, 3.6.6, 3.7.8, 3.7.10, Condition A, Required Action A.1 from 72 hours to 10 days. The CT extension may only be invoked once and remains applicable until Raw Water Pump-3B has been refurbished.

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 by Chandu P. Patel for/
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. 221 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 221
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 January 13, 2005, as supplemented by letters dated February 11, May 6, and June 9, 2005, 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. 221, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Douglas V. Pickett for/
Michael L. Marshall, Jr., 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: September 15, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 221

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

Remove	Insert
3.5-4	3.5-4
3.6-17	3.6-17
3.7-17	3.7-17
3.7-21	3.7-21
B 3.5-15	B 3.5-15
B 3.6-39	B 3.6-39
B 3.6-40	B 3.6-40
B 3.7-44	B 3.7-44
B 3.7-54	B 3.7-54

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 221 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 January 13, 2005, as supplemented by letters dated February 11, May 6, and June 9, 2005, Florida Power Corporation (the licensee also doing business as Progress Energy Florida) requested changes to the Improved Technical Specifications (ITS) for the Crystal River Unit 3 Nuclear Generating Plant (CR-3).

The supplemental letters provided clarifying information that did not expand the scope of the original application or change the initial proposed no significant hazards consideration determination.

1.1 Proposed License Amendment

The proposed change would increase, on a temporary basis, the completion time (CT) to refurbish a Decay Heat Seawater System (raw water (RW)) pump (RWP) that exhibits a degraded flush flow condition. Specifically, the proposed change would revise the Condition A, Required Action A.1, CT for 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,” from 72 hours to 10 days. The CT extension may only be invoked once and remains applicable until RWP-3B has been refurbished.

The requested changes are sought to allow the refurbishment of one RWP (RWP-3B) while the plant is at power. Recent inservice testing shows RWP-3B has been exhibiting a trend of degraded flush flow, and presents the need for this pump’s refurbishment. Since the anticipated duration of the repair activity is greater than the 72-hour CT specified in ITS 3.7.10, the repair can only be performed in MODE 5 or 6 unless the temporary extension of the CT for up to 10 days is approved. Thus, the proposed license amendment would allow the performance of the repair while the plant is at power, and would prevent a forced outage.

1.2 Related U.S. Nuclear Regulatory Commission (NRC) Activities

This license amendment is not related to or in response to any ongoing NRC activities (e.g., generic letters).

2.0 REGULATORY EVALUATION

The staff finds that the licensee, in Attachment B, page 2 of its submittal, identified the applicable regulatory requirements.

2.1 Description of System/Component and Current Requirements

The Decay Heat Seawater System and the Nuclear Services Seawater System comprise the RW system. Seawater is drawn from the intake canal and conveyed to the sump pit via two redundant 48-inch intake conduits (designated “A” and “B”). The "A" intake conduit shares a common intake structure, bar racks, and traveling screens with the Circulating Water System while the “B” intake conduit is supplied with a bar rack and separate traveling screen located in a separate intake structure. The intake conduits are installed individually to one of the two compartments comprising the sump pit. A permanently closed sluice gate separates the two compartments. The seawater pumps, of the vertical wet-pit type, are apportioned in the sump pit as follows:

“A” Compartment:

- One 100% capacity Emergency Nuclear Services Seawater Pump (RWP-2A).
- One 100% capacity Decay Heat Service Seawater Pump (RWP-3A).

“B” Compartment:

- One 100% capacity Normal Nuclear Services Seawater Pump (RWP-1), which is nonsafety related, has a nonseismically qualified motor, has a lower flow capacity than either RWP-2A or RWP-2B, and is not connected to an emergency power source.
- One 100% capacity Emergency Nuclear Services Seawater Pump (RWP-2B).
- One 100% capacity Decay Heat Service Seawater Pump (RWP-3B).

The Nuclear Services Seawater System supplies flow to the Nuclear Services Closed Cycle Cooling System (NSCCC) heat exchangers. The NSCCC, which consists of a single train, supplies the following equipment:

- Motor-driven emergency feedwater pump lube oil cooler and motor cooler
- Nuclear service closed cycle cooling pump motor air coolers
- Nuclear service sea water pump motor air coolers
- Spent fuel coolers
- Spent fuel coolant pumps air handling units
- Steam generator sample coolers and pressurizer sample cooler
- Control rod drive mechanism stator water jacket coolers with increased head provided by the nuclear service booster pumps
- Letdown coolers
- Reactor coolant drain tank cooler
- Seal return coolers
- Waste gas compressors
- Makeup and purification pumps and motors: 1A (normal source), 1B (only source), and 1C (backup source)

- Reactor coolant pump motor upper and lower bearing coolers
- Reactor coolant pump motor air coolers
- Reactor coolant pump seal heat exchangers
- Post accident sample precooler and post accident sample coolers
- RB fan assembly cooling coils
- RB ventilation fan motor coolers
- Chilled water system chillers

The Decay Heat Seawater System supplies flow to the Decay Heat Closed Cycle Cooling System. It contains two separate and independent trains that supply the following equipment:

- Decay heat removal heat exchangers
- Decay heat service sea water pump motors
- Decay heat closed cycle cooling water pump motor air handling units
- Decay heat pumps and motors
- Reactor Building spray pumps and motors
- Makeup and purification pumps and motors: 1A (backup source from Train “A”) and 1C (normal source from Train “B”)

Seawater is circulated through the nuclear services heat exchangers and merged with the seawater from the Decay Heat Closed Cycle System heat exchangers to the redundant 48-inch discharge pipes leading to the discharge canal. Three of the four nuclear services heat exchangers supply the full normal and emergency cooling requirements, with the fourth unit on reserve.

Informal calculations performed by the licensee suggest that below an ultimate heat sink temperature of 90 °F, RWP-1 can provide enough flow to remove heat loads in accident conditions.

2.2 Applicable Regulatory Criteria/Guidelines

The regulatory criteria/guidelines on which the staff based 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, 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. 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 regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.

One acceptable approach to 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 provide numerical risk acceptance guidelines for evaluating temporary TS changes against the fourth key principle. It can be demonstrated with reasonable assurance that a temporary TS change meets the fourth key principle 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 staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in Attachments A and B of the licensee's submittal. The detailed evaluation described in this section supports the conclusion 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

3.1 Detailed Description of the Proposed Change

The current CT for ITS 3.5.5, 3.6.6, 3.7.8, and 3.7.10, Condition A, Required Action A.1 is 72 hours.

Emergency Core Cooling System

The proposed amendment adds a note to ITS 3.5.5, Condition A, Required Action A.1 that would increase the CT from 72 hours to 10 days. The proposed note states the following:

> *On a one-time basis, an Emergency Core Cooling System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

The ITS Bases for 3.5.5, Required Action A.1 will be revised by adding a footnote as follows:

> *On a one-time basis, an Emergency Core Cooling System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

Reactor Building Spray System

The proposed amendment adds a note to ITS 3.6.6, Condition A, Required Action A.1 that would increase the CT from 72 hours to 10 days. The proposed note states the following:

> *On a one-time basis, a Reactor Building Spray System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

The ITS Bases for 3.6.6, Required Action A.1 will be revised by adding a footnote as follows:

> *On a one-time basis, a Reactor Building Spray System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

Decay Heat Closed Cycle Cooling Water System

The proposed amendment adds a note to ITS 3.7.8, Condition A, Required Action A.1 that would increase the CT from 72 hours to 10 days. The proposed note states the following:

> *On a one-time basis, a Decay Heat Closed Cycle Cooling Water System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

The ITS Bases for 3.7.8, Required Action A.1 will be revised by adding a footnote as follows:

> *On a one-time basis, a Decay Heat Closed Cycle Cooling Water System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

<u>Decay Heat Seawater System</u>

The proposed amendment adds a note to ITS 3.7.10, Condition A, Required Action A.1 that would increase the CT from 72 hours to 10 days. The proposed note states the following:

> *On a one-time basis, a Decay Heat Seawater System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

In addition, the ITS Bases for 3.7.10, Action A.1 will be revised as follows:

> *On a one-time basis, a Decay Heat Closed Cycle Cooling Water System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.

3.2 Staff Review Methodology

As required by the Standard Review Plan (SRP), Chapter 16.1, "Risk-Informed Decisionmaking: Technical Specifications," the staff reviewed the submittal against the five key principles of the staff's philosophy of risk-informed decisionmaking listed in RG 1.177, Section B.

3.3 Key Information Used in Staff Review

The key information used in the staff's review of the risk evaluation is contained in Attachments A and E to the licensee's submittal (Reference 1), as supplemented by the licensee in response to the staff's request for additional information (Reference 2), and the licensee's revised amendment request (Reference 3). In addition, the staff consulted the safety evaluation reports (SERs) on the individual plant examinations (IPEs) and individual plant examinations - external events (IPEEEs) submitted by the licensee (References 4 and 5).

3.4 Comparison Against Regulatory Criteria/Guidelines

The staff's comparison of the licensee's proposed license amendment for a temporary extension of the Decay Heat Seawater System train CT against the five key principles is presented in the following sections.

3.4.1 Traditional Engineering Evaluation

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

3.4.1.1 Compliance with Current Regulations

Design basis analyses are not impacted by the proposed change and consequently, safety margins are not affected. 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. Therefore, based on traditional engineering considerations, the NRC staff considers the proposed one-time TS change to be acceptable.

3.4.1.2 Evaluation of Defense-in-Depth

The staff observes that there will be sufficient diverse means of ensuring core cooling during the refurbishment of RWP-3B. Specifically:

- The Emergency Feedwater System (the motor-driven pump is cooled by NSCCC, and the turbine-driven and diesel-driven pumps are self-cooled)
- The Emergency Diesel Generators (self-cooled)
- The Makeup and Purification Pumps provide the high-pressure injection function (which will be aligned among the NSCCC and Decay Heat Seawater System Train "A" supplied by RWP-3A)
- One decay heat removal heat exchanger
- One decay heat service seawater pump and associated air handling unit
- One decay heat pump, which provides the low-pressure injection function
- One Reactor Building spray pump

As discussed elsewhere in this evaluation, the licensee has established compensatory measures and license commitments to better assure the capability of a single Decay Heat Closed Cycle Cooling Water System train to function during the proposed temporary extended outage time of the other train while it is being refurbished. The NRC staff considers the licensee's actions to be appropriate and adequate for maintaining defense-in-depth during the extended outage period.

3.4.1.3 Evaluation of Safety Margins

Design basis analyses are not impacted by the proposed change and consequently, safety margins are not affected.

3.4.2 Risk Evaluation

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

- Tier 1 - The first tier evaluates the licensee's probabilistic risk assessment (PRA) and the impact of the change on plant operational risk, as expressed by the change in core damage frequency (CDF) and the change in large early release frequency (LERF). The change in risk is compared against the acceptance guidelines presented in RG 1.174. The first tier also aims to ensure that plant risk does not increase unacceptably during the period when equipment is taken out of service per the license amendment, as expressed by the incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). The incremental risk is compared against the acceptance guidelines presented in RG 1.177.

- 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 out of service 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 extension 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 CRMP is to ensure that equipment removed from service prior to or during the proposed extended CT period will be appropriately assessed from a risk perspective.

3.4.2.1 Tier 1: PRA Capability and Insights

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

3.4.2.1.1 Evaluation of PRA Technical Adequacy

To determine whether the PRA used in support of the proposed CT extension is of sufficient quality, scope, and level of detail, the staff evaluated the relevant information provided by the licensee in their submittal, as supplemented, and considered the findings of recent PRA reviews. The staff's review of the licensee's submittal focused on the validity of the licensee's PRA model to analyze the risks stemming from the proposed CT extension and did not involve an in-depth review of the licensee's PRA.

The PRA used to support the licensee’s submittal is a revision and extension of the original Level 1 PRA study completed in 1987, which was submitted to the staff and reviewed by Argonne National Laboratory in NUREG/CR-5245. This original work, which addressed internal initiating events, was revised, augmented to include internal floods and a limited scope Level 2 PRA, 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 PRA 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 - 10 CFR 50.54(f).”

Dates when the licensee submitted its IPE and IPEEE, and the corresponding dates when the staff issued its SER, are given below:

Analysis	Date of Submittal	SE Issued
IPE	March 9, 1993	June 30, 1998
IPEEE	June 28, 1996	January 11, 2001

Subsequent revisions to the PRA models have been performed by qualified individuals with knowledge of PRA methods 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 PRA models have been updated for various reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, and advances in PRA technology.

The CR-3 PRA model and documentation was subjected to the industry peer certification review process in September 2001. In preparation for this 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 failures analyses were updated to current industry methodologies and data sources. An internal review of the PRA model elements and their corresponding documentation was conducted to assure that the model and documentation reflected the plant design.

The industry peer certification review was conducted by a diverse group of PRA engineers from other Babcock and Wilcox (B&W) plants, industry PRA consultants familiar with the B&W plant design, and a representative from the Institute of Nuclear Power Operations. The certification review covered all aspects of the internal events PRA 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 PRA model, and enhancements to the documentation of the model and the administrative procedures used for model updates.

Issues involving model documentation are being addressed as each individual PRA document is reviewed and approved under Progress Energy corporate procedures. Other changes involving guidance documents and administrative processes used for model updates are planned to be addressed by Progress Energy corporate procedures once the peer review process has been completed for all PRA models (including the Brunswick Nuclear Plant, the Robinson Nuclear Plant, and the Harris Nuclear Plant). According to the licensee, the issues identified by the peer review in these areas have been reviewed and determined not to have any impact on the present submittal. All other peer review items that impact the PRA model have been addressed and are reflected in the present submittal according to the licensee.

In 2003, the staff assessed the quality of the CR-3 PRA used in support of a license amendment to extend the CT of the emergency diesel generators. The safety evaluation (SE) attached to the issuance of the license amendment, issued June 13, 2003, indicated that the risk analysis used in support of the license amendment was of sufficient quality for that application.

In 2004, the staff assessed the quality of the CR-3 PRA used in support of a license amendment to temporarily extend the CT of RWP-2A and RWP-2B. The SE attached to the issuance of the license amendment, issued May 18, 2004, indicated the risk analysis used in support of the license amendment was of sufficient quality for that application, which is essentially the same scope as the current application.

Based on review of the above information, the 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 (Section 2.2.3 and 2.5), and SRP Chapter 19.1, and that the quality of the CR-3 , Unit 3 PRA is sufficient to support the risk evaluation provided by the licensee in the proposed license amendment.

3.4.2.1.2 Evaluation of PRA Results and Insights

As previously discussed, satisfaction of the fourth key principle of risk-informed decisionmaking may be demonstrated with reasonable assurance by comparing risk metrics that reflect the proposed TS change to the numerical risk acceptance guidelines in RG 1.174 and RG 1.177. Attachment B of the licensee's letter dated June 5, 2005 (Reference 3), provides relevant risk calculations, and are based on the following assumptions:

1. The power-operated relief valve block valve (RCV-11) is closed, which is consistent with the expected plant configuration that will exist when RWP-3B is repaired. The licensee has stated that RCV-11 was closed in order to isolate a reactor coolant system to reactor building atmospheric leak (approximately 2.5 gallons per minute), which was discovered on March 3, 2005, following the quarterly stroke test of RSV-11. CR-3 will be operated with RCV-11 closed until Refueling Outage 14, which is scheduled for fall 2005.

2. The plant configuration prior to an accident, which impacts the PRA results, is the historically preferred configuration identified below:

   a. Makeup pump MUP-1B is running, powered from the engineered safeguards (ES) 4160 "A" bus, and ES selected

   b. Makeup pump MUP-1C is in ES standby and cooled from the Decay Heat Closed Cycle Cooling System, which is cooled by the Decay Heat Seawater System

   c. Makeup pump MUP-1A is not ES selected, but available and cooled from NSCCC, which is cooled by the Nuclear Services Seawater System.

   d. ES 4160 "A" is powered from the offsite power transformer

   e. ES 4160 "B" is powered from the backup engineered safeguards transformer

   f. RWP-1 and SWP-1C are the normally running cooling water pumps

For accidents initiated by internal events (including internal floods), the licensee used its model of record (MOR) PRA, which is an "average maintenance" PRA model, to compute the risk metrics needed for comparison against the numerical risk acceptance guidelines in RG 1.174. The following results were provided by the licensee:

Risk Metric	Baseline (per year)	Change (per year)
CDF	5.4×10^{-6}	4.0×10^{-7}
LERF	4.0×10^{-7}	below 10^{-9}

The licensee used the CR-3 equipment out-of-service (EOOS) model, which is a "zero maintenance" PRA model, to compute the risk metrics needed for comparison against the numerical risk acceptance guidelines in RG 1.177. The licensee determined that the ICCDP associated with the proposed change was 1.2×10^{-6} and that the ICLERP was 2.6×10^{-9}.

The staff observes that, ideally, all risk metrics should be determined by adjusting an "average" PRA model (i.e., a PRA model that includes contributions from equipment maintenance unavailability). Use of a "zero maintenance" PRA model, which omits maintenance unavailability contributions, to determine the RG 1.174 and RG 1.177 risk metrics introduces additional uncertainty into the analysis. However, the contribution from equipment maintenance unavailability to changes in risk depends on the likelihood of performing maintenance on other plant equipment in parallel with maintenance on the equipment whose CT is being extended. The likelihood of simultaneous maintenance actions is judged to be small and will be controlled by the licensee by its risk-informed configuration management programs (discussed in Section 3.4.2.3 below). Therefore, the magnitude of the additional uncertainty resulting from use of a "zero maintenance" PRA model is small. The staff concludes that the licensee's use of a "zero maintenance" PRA model is acceptable for evaluating the risk metrics referenced in RG 1.177.

In order to assess the impact of the proposed change on the risk of internal fires, the licensee identified fire zones that contain circuits applicable to RWP-3A and the front line systems that it supports (decay heat removal and decay heat closed cycle cooling). The greatest risk impact due to RWP-3B being out of service is expected for fires that impact only the "A" equipment trains. The licensee then estimated the instantaneous CDF due to internal fires when RWP-3B is out of service. This estimate was based on a review of information contained in the IPEEE. Fires initiated by transient combustibles were screened out for each fire zone; in addition, fires initiated by equipment that will not be operated without special precautions were screened out. Credit was taken for automatic fire suppression capability as appropriate. A value of 0.1 was assumed for the conditional core-damage probability given that a fire occurs. A summary of the estimate is presented in the following table.

Fire Zone	Contains Fires Sources Other than Transient Combustibles and Equipment That Will Not Be Operated Without Special Precautions	Credit for Automatic Suppression	CDF During RWP-3B Refurbishment
AB-75-5	no	no	0
AB-95-3AA	yes	yes	5.21×10^{-7}
AB-95-3C	yes	yes	8.96×10^{-8}
AB-95-3E	no	yes	0
AB-95-3F	no	no	0
AB-95-3K	no	no	0
AB-95-3L	no	no	0
AB-95-3M	no	no	0
AB-95-3N	no	no	0
AB-95-3P	no	no	0
AB-95-3Q	no	no	0
AB-95-3R	yes	no	1.21×10^{-5}
AB-95-3T	no	no	0
AB-95-3U	no	no	0
AB-95-3W	no	no	0
AB-119-6A	yes	yes	8.00×10^{-6}
AB-119-6E	yes	yes	8.76×10^{-6}
CC-108-102	yes	no	2.30×10^{-6}
CC-108-104	no	no	0
CC-108-106	yes	no	2.71×10^{-5}
CC-108-108	yes	no	1.63×10^{-5}
CC-108-110	yes	no	9.28×10^{-6}
CC-124-111	yes	yes	4.09×10^{-6}
CC-124-117	yes	no	1.07×10^{-5}
		TOTAL	9.92×10^{-5}

Based on this analysis, the licensee estimated that the ICCDP due to internal fires was 2.7×10^{-6}.

The licensee stated that it does not have a fire PRA model that can be used to quantify the effect of postulated fire scenarios on LERF. The significant contributors to LERF involve containment bypass sequences (steam generator tube ruptures (SGTR), and interfacing systems loss-of-coolant accident (ISLOCA). The fire-related LERF impact of the proposed changes is estimated to be very small because (a) a fire in the “A” 4160V switchgear room would not increase the frequency of these initiating events, and (b) the likelihood of a fire in the switchgear room coincident with an SGTR or ISLOCA is very small.

RG 1.177, Section 2.3.2 states that the scope of the risk evaluations made to assess changes to TS requirements should include internal fires. Ideally, quantitative evaluations should be made; however, qualitative arguments, bounding analyses, and compensatory measures may also be used. The staff concludes that the licensee’s assessment of the risks due to internal fires during RWP-3B refurbishment is acceptable because:

- The identification of significant fire zones during RWP-3B refurbishment was based on generally accepted industry good practices.

- The licensee’s estimate of the CCDP given a fire is conservative since there are several diverse means of providing core cooling if the Decay Heat Seawater System is not available due a fire that disables RWP-3A while RWP-3B is being refurbished.

- The licensee will establish fire watches in fire zones containing circuits to the RWP-3A and RWP-3B pumps (refer to the list of regulatory commitments in Section 4).

For the seismic risk analysis, 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 of 0.1g peak ground acceleration. The seismic IPEEE evaluation took credit for plant modifications and activities that had been identified under the Unresolved Safety Issue (USI) A-46 program, but were not yet implemented when the IPEEE was submitted. The credited plant modifications and activities were subsequently implemented, and the USI A-46 program was closed out in August 2000. Since the seismic margins approach was used, no quantitative estimate was made for the seismic contribution to plant CDF. The staff concludes that the seismic risk during RWP-3B refurbishment is acceptable because the CR-3 site is located in a region of low seismicity.

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. CR-3 did not quantitatively estimate the contribution to CDF from HFO events since these events were screened out on the basis of low occurrence frequency using the NUREG-1407 screening approach. CR-3 performed walkdowns to confirm that no plant changes had occurred since the plant was licensed that would impact on the IPEEE review. The staff concludes that the risk from HFO events during RWP-3B refurbishment is acceptable because:

- The licensee will not initiate an extended RWP-3B maintenance outage if adverse weather, as designated by the Emergency Preparedness procedures, is anticipated (refer to the list of regulatory commitments in Section 4).

- The staff previously accepted the licensee's HFO event risk screening approach during its review of the licensee's IPEEE submittal.

Section 2.4 of RG 1.177 states that a permanent TS CT change has only a small quantitative impact on plant risk if the ICCDP is less than $5x10^{-7}$ and the ICLERP is less than $5.x10^{-8}$. The ICCDP value associated with RWP-3B replacement, 3.9×10^{-6} (determined by summing the contributions from the internal initiating events and internal fires) is above the RG 1.177 risk acceptance guidelines. The staff concludes that the risk impact of the proposed change is acceptable for the following reasons:

- The proposed license amendment concerns a temporary change to the technical specifications. As previously noted, RG 1.177 is directly applicable only to permanent changes to TS requirements.

- The licensee's estimate of the fire risk contribution is conservative. A more realistic calculation would result in smaller ICCDP values.

- The licensee has proposed compensatory measures (refer to the Tier 2 evaluation), particularly measures to minimize the fire-related risks, during the planned refurbishment activities.

- The ICCDP value for the proposed change is similar to the value accepted by the staff in the SE of the license amendment to temporarily extend the CT of RWP-2A and RWP-2B, which was issued May 18, 2004.

Section 2.4 of RG 1.177 requires the comparison of risk metrics to the risk acceptance guidelines contained in Section 2.2.4 (ΔCDF versus baseline CDF) and Section 2.2.5 (ΔLERF versus baseline LERF) of RG 1.174. Based upon information provided by the licensee and considering the conservatisms and uncertainties in the analysis, the staff concludes that the proposed change results in an acceptable increase in risk that is 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.

3.4.2.2 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, are taken out of 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 RWP refurbishment activities, the license reviewed its PRA results to identify compensatory measures that minimize risk. These compensatory measures include:

- CR-3 will perform compliance procedure CP-253, "Power Operation Risk Assessment and Management," which requires a deterministic and probabilistic evaluation of risk for the performance of all activities.

- CR-3 will select beneficial makeup pump configurations.

- Operator attention to the importance of protecting the operable redundant train and support systems will be increased.

- Operator attention to non-safety grade FWP-7 and standby diesel generator (MTDG-1) will be increased. This will be accomplished by on-shift operating crew review of Emergency Operation Procedure (EOP)-14, Enclosure 7, "Emergency Feedwater Pump (EFWP) Management."

- CR-3 will not schedule elective maintenance in the switchyard that would challenge the availability of offsite power.

- CR-3 will establish fire watches, as required, in fire zones identified as containing circuits applicable to the RWP-3A and RWP-3B pumps to minimize fire risk in these areas.

- CR-3 will not initiate an extended RWP-3B maintenance outage if adverse weather, as designated by the Emergency Preparedness procedures, is anticipated.

- CR-3 will evaluate the material condition of the redundant train to ensure that there is no negative trend that could challenge operability.

The review of PRA results to identify compensatory measures demonstrates the licensee's ability to recognize and avoid risk-significant plant configurations. Therefore, the 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.

3.4.2.3 Tier 3: Risk-Informed Configuration Risk Management

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

To ensure that defense-in-depth capabilities and the assumptions in the PRA are maintained during the proposed temporary extended CT, the licensee will continue the performance of 10 CFR 50.65(a)(4) assessments before performing maintenance or surveillance activities. In addition, no maintenance activities of risk sensitive equipment beyond that required for the RWP refurbishment activity will be concurrently scheduled. The staff notes that the CR-3 EOOS computer model provides the licensee with real-time risk monitoring capability.

Therefore, the 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.

4.0 <u>REGULATORY COMMITMENTS</u>

The licensee has agreed to the following regulatory commitments, which apply during refurbishment of Decay Heat Seawater System Pump RWP-3B:

20. CR-3 will perform compliance procedure CP-253, “Power Operation Risk Assessment and Management,” which requires a deterministic and probabilistic evaluation of risk for the performance of all activities.

21. CR-3 will select beneficial makeup pump configurations.

22. Operator attention to the importance of protecting the operable redundant train and support systems will be increased.

23. Operator attention to non-safety grade FWP-7 and standby diesel generator (MTDG-1) will be increased. This will be accomplished by on-shift operating crew review of EOP-14, Enclosure 7, “Emergency Feedwater Pump (EFWP) Management.”

24. CR-3 will not schedule elective maintenance in the switchyard that would challenge the availability of offsite power.

25. CR-3 will establish fire watches, as required, in fire zones identified as containing circuits applicable to the RWP-3A and RWP-3B pumps to minimize fire risk in these areas.

26. CR-3 will not initiate an extended RWP-3B maintenance outage if adverse weather, as designated by the Emergency Preparedness procedures, is anticipated.

27. CR-3 will evaluate the material condition of the redundant train to ensure that there is no negative trend that could challenge operability.

9. Equipment and systems (including support systems) will be designated “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.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitment(s) are best 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 CONSIDERATION

The amendment changes surveillance requirements. 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 (70 FR 5246). 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 staff finds that the licensee's proposed change to revise, on a temporary basis, the CT for ITS 3.5.2, 3.6.6, 3.7.8, and 3.7.10 from 72 hours to 10 days is acceptable because the five key principles of risk-informed decisionmaking identified in RG 1.174 and RG 1.177 have been satisfied.

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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

28. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, “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,” 3F0105-02, January 13, 2005.

29. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, “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 (TAC No. MC5631),” 3F0505-07, May 6, 2005.

30. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, “Crystal River, Unit 3 - License Amendment Request #289, Revision 1 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,” 3F0605-01, June 9, 2005.

31. Letter from U.S. Nuclear Regulatory Commission to John Paul Cowan, Florida Power Corporation, “Crystal River, Unit 3 - Supplemental Staff Evaluation Report Regarding Individual Plant Examination Report - Internal Events (TAC No. M74401),” June 30, 1998.

32. Letter from U.S. Nuclear Regulatory Commission to Dale E. Young, Crystal River Nuclear Plant, “Review of Crystal River Unit 3 Individual Plant Examination of External Events (IPEEE) Submittal, (TAC No. M83612),” January 11, 2001.

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