May 18, 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 THE NUCLEAR SERVICES SEAWATER SYSTEM (TAC NO. MC0119)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 212 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 July 14, 2003, as supplemented by letters dated November 20, 2003, March 25, 2004, and April 27, 2004. The proposed changes would revise TS 3.7.9 to allow a one-time increase in the Completion Time (CT) for restoring an inoperable nuclear services seawater system train to operable status. Specifically, the proposed change would revise the CT for TS 3.7.9, Condition A, Required Action A.1, from 72 hours to 10 days. The CT extension may only be invoked once and remains applicable until December 30, 2004.

In addition, the amendment adds License Condition 2.C.(13). During the one-time extended allowed outage time for work on nuclear services seawater system emergency pump RWP-2B, authorized in TS 3.7.9, Florida Power Corporation will implement compensatory measures for CR-3 as described in the license condition.

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. 212 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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In addition, the amendment adds License Condition 2.C.(13). During the one-time extended allowed outage time for work on nuclear services seawater system emergency pump RWP-2B, authorized in TS 3.7.9, Florida Power Corporation will implement compensatory measures for CR-3 as described in the license condition.

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cc w/enclosures: See next page

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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. 212
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 July 14, 2003, as supplemented November 20, 2003, March 25, 2004, and April 27, 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. 212, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. The license is also amended by the addition of paragraph 2.C.(13), which reads as follows:

    2.C.(13) During the one-time extended allowed outage time for work on Nuclear Services Seawater System Emergency Pump RWP-2B, authorized in Improved Technical Specification (ITS) 3.7.9, CR-3 will implement the following compensatory measures:

    (a) No elective maintenance will be scheduled on other related risk sensitive equipment beyond that required for the refurbishment activity of RWP-2B that could degrade the risk profile of the plant. For these purposes, the systems are: Nuclear Services and Decay Heat Seawater System, Decay Heat System, Decay Heat Closed Cycle Cooling Water System, Nuclear Services Closed Cycle Cooling Water (SW), Emergency Diesel Generators, Chilled Water, Emergency Feedwater System, Emergency Feedwater Initiation and Controls System (EFIC) Auxiliary Feedwater Pump and Makeup System.

    (b) Makeup Pump configurations will be selected to minimize risk.

    (c) Increased operator attention will be focused on loss/restoration of RW/SW/ redundant train and the Appendix R Chiller. This will be accomplished by on shift operating crew review of Abnormal Procedure (AP)-330, Loss of Nuclear Service Cooling.

    (d) Operator attention will be focused on potential use of non-safety grade Feedwater Pump FWP-7 and its dedicated Diesel Generator (MTDG-1). This will be accomplished by on shift operating crew review of Emergency Operating Procedure (EOP)-14, Enclosure 7, Emergency Feedwater Pump (EFWP) Management.

    (e) Daily operator walkdowns of the redundant train of RW/SW/Pumps and associated power supply switchgear will be conducted.

    (f) No elective maintenance to be scheduled in the switchyard that would challenge the availability of offsite power to the ES Buses or to the Bus for RWP-1.

(g) Hourly roving fire watches will be established in fire zones identified as containing circuits required for RWP-1 or RWP-2A to minimize fire risk in these areas. Those fire zones are: AB-95-3AA, AB-95-3B, AB-95-3E, AB-95-3F, AB-95-3G, AB-95-3K, AB-95-3T, AB-95-3U, AB-95-3W, AB-95-3X, AB-95-3Z, CC-108-102, CC-108-104, CC-108-105, CC-108-106, CC-108-108, CC-108-109, CC-108-110, CC-124-111, CC-124-116, CC-124-117, CC-134-118A, IB-95-200C, IB-119-201B, TB-119-400E, TB-119-403, TB-95-400A, TB-95-401, AB-119-6 and AB-119-6A.

(h) Prior to entering the Allowed Outage Time for rebuild of RWP-2B, fire zones containing circuits for either RWP-1 or RWP-2A will be walked down to identify and minimize transient combustibles not related to ongoing approved work. Hourly roving fire watches will be charged with continuing to monitor for the presence of transient combustible materials until RWP-2B is returned to service.

(i) For the risk significant fire zones containing circuits for both RWP-1 and RWP-2A, the following additional compensatory measures will be established which address the specific risk factors in each zone. No additional compensatory measures beyond roving fire watches will be established for fire zones that contain circuits for all three pumps (RWP-1, RWP-2A and RWP-2B).

<u>Fire Zone CC-108-108</u>
A continuous Fire Brigade Qualified fire watch will be stationed in the fire area except while 4160V breaker manipulations are being performed. The individual's turnout gear will be available in the adjacent area where a fire hose station is also located equipped with an electrically safe fog nozzle.

<u>Fire Zone AB-95-3W</u>
The Waste Transfer Pumps will only be operated when there is a qualified fire watch in the immediate vicinity of the operating pump equipped with a radio and fire extinguisher. No hot or spark producing work will be conducted. A roving hourly fire watch will observe this zone.

<u>Fire Zone AB-95-3E</u>
The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.

<u>Fire Zone AB-95-3AA</u>
The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.

Fire Zone AB-95-3F
The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.

Fire Zone AB-95-3T
The portion of the fire zone that can be locked will be locked and the keys will be administratively controlled. Entries will be limited to only operationally necessary activities and require inspection for transient combustible materials upon exit. The portion of the zone that cannot be locked will be observed by the roving fire watch. No hot or spark producing work will be conducted.

Fire Zone AB-95-3U
The portion of the fire zone that can be locked will be locked and the keys will be administratively controlled. Entries will be limited to only operationally necessary activities and require inspection for transient combustible materials upon exit. The portion of the zone that cannot be locked will be observed by the roving fire watch. No hot or spark producing work will be conducted.

4. 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

/RA/

William F. Burton, Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments:
1. Changes to License pages
2. Changes to the Technical Specifications

Date of Issuance: May 18, 2004

ATTACHMENT 1 TO LICENSE AMENDMENT NO. 212

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Revised License No. DPR-72 as follows:

Remove	Insert
5a	5a through 5c

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 212

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
3.7-19	3.7-19
B 3.7-48	B 3.7-48

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 212 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 July 14, 2003, as supplemented by letters dated November 20, 2003, March 25, 2004, and April 27, 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 November 20, 2003, March 25, 2004, and April 27, 2004, supplemental letters 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.

1.1 Proposed License Amendment

The proposed change would increase, on a one-time basis, the Completion Time (CT) to restore an inoperable Nuclear Services Seawater System train. Specifically, the proposed change would revise the CT for TS 3.7.9, Condition A, Required Action A.1, from 72 hours to 10 days. The CT extension may only be invoked once and remains applicable until December 30, 2004.

The requested changes would allow the refurbishment of one Nuclear Services Seawater System Emergency Pump (RWP-2A or RWP-2B) online. Recent in-service testing indicates declining performance of the RWP-2A and RWP-2B pumps, and presents the need for their refurbishment. RWP-2A has been exhibiting a trend of decreasing pump differential pressure (dP). RWP-2B is in the "Alert Range" for vibration. It is impractical to refurbish both pumps during a refueling outage since there is only one spare rotating unit available, and there is not sufficient time for refurbishing the replaced rotating unit and installing it on the remaining pump during the current CT outage. Since the anticipated duration of the repair activity is greater than the 72-hour CT specified in ITS 3.7.9, the repair can only be performed in MODE 5 or 6 unless the one-time extension of the CT for up to 10 days is approved. Thus, the proposed license amendment would allow the performance of the repair online and would prevent a forced outage.

1.2 Related 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 NRC staff finds that the licensee's reference to the maintenance rule, 10 CFR 50.65 (in Attachment B, Page 2 of its July 14, 2003, submittal) identified the regulatory requirements applicable to this license amendment request.

2.1 Description of System/Component and Current Requirements

The decay heat seawater system and the nuclear services seawater system comprise the raw water (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 (CW) 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 non-safety related, has a non-seismically 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 (Service Water) system heat exchangers. The decay heat seawater system supplies flow to the decay heat closed cycle cooling system. These systems cool systems, structures, and components (SSCs) that are relied upon for accident mitigation such as the control complex chillers and the reactor building cooling units.

Seawater is circulated through the nuclear services heat exchangers and merged with the seawater from the decay heat closed cycle heat exchangers to the redundant 48-inch discharge pipes leading to the discharge canal. Three of the four nuclear service 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 (UHS) temperature of 90 °F, RWP-1 can provide enough flow to remove heat loads in accident conditions. In order to remove or install the motor on RWP-2A, it is necessary to temporarily remove the RWP-3A header piping.

2.2 Applicable Regulatory Criteria/Guidelines

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

- Regulatory Guide (RG) 1.174 (RG 1.174), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The RG 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." The RG 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. RG 1.177 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 RG provides 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 NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, as 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 3.7.9, Condition A, Required Action A.1 is 72 hours.

The proposed amendment adds a note to ITS 3.7.9, 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 Nuclear Services Seawater System train may be inoperable for up to 10 days to allow performance of Nuclear Services Seawater Emergency Pump RWP-2A or RWP-2B repairs. The ability to apply the 10-day Completion Time will expire on December 30, 2004.

Consistent with the TS, the ITS Bases for 3.7.9, Action A.1 are revised as follows:

> With one of the Nuclear Services Seawater pumps inoperable, action must be taken to restore the pump to OPERABLE status within 72* hours....
>
> *On a one-time basis, a Nuclear Services Seawater Services train may be inoperable for up to 10 days to allow performance of Nuclear Services Seawater Emergency Pump RWP-2A or RWP-2B repairs. The ability to apply the 10-day Completion Time will expire on December 30, 2004.

3.2 NRC Staff Review Methodology

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

3.3 Key Information Used in NRC staff Review

The key information used in the NRC staff's review of the risk evaluation is contained in Attachments A and E to the licensee's submittal dated July 14, 2003, as supplemented by the licensee's letters dated November 20, 2003, March 25, 2004, and April 27, 2004. In addition, the NRC staff consulted the NRC staff's Safety Evaluations (SEs) on the Individual Plant Examinations (IPEs) and Individual Plant Examinations - External Events (IPEEEs) submitted by the licensee on June 30, 1998, and January 11, 2001, respectively.

3.4 Comparison Against Regulatory Criteria/Guidelines

The NRC staff's comparison of the proposed license amendment for a one-time extension of the nuclear services 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 of risk-informed changes to TS requirements includes consideration of defense-in-depth, safety margins, and compliance with existing regulations. These considerations apply more directly to permanent changes to TS requirements, but are also considered to some extent for temporary changes that are proposed. As discussed in Sections 3.4.2 and 4.0 of this evaluation, the licensee has established compensatory measures and license conditions to better assure the capability of a single nuclear services seawater system train to function during the proposed one-time 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. 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.

3.4.2 Risk Evaluation

The risk evaluation presented below addresses the last two key principles of the NRC staff's philosophy of risk-informed decision making, which concern 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 RG 1.177.

- Tier 1 - The first tier evaluates the licensee's probabilistic risk/safety assessment (PRA/PSA) 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 AOT 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 AOT period will be appropriately assessed from a risk perspective.

3.4.2.1 Tier 1: PRA Capability and Insights

The Tier 1 NRC staff review involved two aspects: (1) evaluation of the validity 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 Validity

To determine whether the PRA used in support of the proposed CT extension is of sufficient quality, scope, and level of detail, the NRC staff evaluated the relevant information provided by the licensee in their submittal, as supplemented, and considered the findings of recent PRA reviews. The NRC 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 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 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 NRC staff issued its SE, are given below:

Analysis	Date of Submittal	Date When SER 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 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 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 the licensee's corporate procedures. Other changes involving guidance documents and administrative processes used for model updates are planned to be addressed by the licensee's 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 NRC staff assessed the quality of the CR-3 PRA used in support of a license amendment to extend the CT of the EDGs. The SE for License Amendment #207, issued June 13, 2003, indicates that the risk analysis used in support of the license amendment was of sufficient quality.

Based on 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 (Section 2.2.3 and 2.5), and SRP Chapter 19.1, and that the quality of the CR-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

Attachment E of the licensee's amendment request provides relevant risk calculations. The CDF and LERF associated with specific plant configurations during RW pump refurbishment were determined using the CR-3 equipment out-of-service (EOOS) model, which is a "zero maintenance" PRA model that is based on the licensee's PRA model of record (MOR) and is used to support a real-time risk monitor. As previously discussed, satisfaction of the fourth key principle of risk-informed decision making 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. Ideally, these 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 that omits maintenance unavailability contributions to determine the RG 1.174 and RG 1.177 risk metrics introduces additional uncertainty into the analysis.

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, and industry-wide PRA results indicate that this likelihood is small. Therefore, the magnitude of the additional uncertainty resulting from use of a "zero maintenance " PRA model is small. The NRC staff concludes that the licensee's use of a "zero maintenance" PRA model is acceptable for evaluating the proposed one-time RW pump CT extension.

The plant configuration prior to an accident impacts the PRA results. The assumed "normal" or historically preferred configuration is:

- Makeup pump MUP-1B is running, powered from the ES 4160 "B" bus, and ES selected
- Makeup pump MUP-1C is in ES standby and cooled from decay heat closed cycle cooling (DHCCC), which is cooled by the decay heat seawater system
- Makeup pump MUP-1A is not ES selected, but available and cooled from nuclear services closed cycle cooling (NSCCC), which is cooled by the nuclear services seawater system.
- ES 4160 "A" is powered from the offsite power transformer (OPT)
- ES 4160 "B" is powered from the backup engineered safeguards transformer (BEST)
- RWP-1 and SWP-1C are the normally running cooling water pumps

The licensee conducted additional sensitivity analyses to determine the impact of the initial plant configuration on plant risk, which indicate that the historically preferred plant configuration listed above is the most limiting with respect to risk.

For the normal plant configuration, the CDF estimated using the baseline (zero maintenance) EOOS model is 2.6E-6/y, based on a truncation frequency of 1E-8/y. The EOOS model regenerates cut sets for specific plant configurations; it does not requantify CDF using presolved cut sets, an approach whose validity depends on the truncation value used. The licensee conducted sensitivity analyses to confirm the adequacy of the selected truncation frequency.

The licensee has estimated the expected duration of the RWP-2A refurbishment to be about 120 hours; during this time, both RWP-2A and RWP-3A are expected to be unavailable about 60 hours. The RWP-3A header piping will be removed twice, the first time to allow removal of the RWP-2A motor and the second time to allow re-installation of the RWP-2A motor.

TS 3.7.10, Condition A, Required Action A.1 specifies a 72-hour CT for RWP-3A. The refurbishment of RWP-2B has an estimated duration of about 72 hours, and does not affect the availability of the other RW pumps.

A review of the licensee's IPEEE results indicates that internal fires are significant contributors to the overall CDF. 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 having 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 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. Because 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. Therefore, the NRC staff concludes that the overall CDF is reasonably approximated by including the contributions from internal initiating events and internal fires.

However, the licensee did not include the contribution from fires in the risk calculations submitted in support of the license amendment request. The NRC staff notes that the planned RW pump refurbishment activities may degrade the plant's capability to prevent or mitigate an accident depending on which RW pump is being refurbished and what equipment is damaged by a fire. The NRC staff estimated the total fire-related CDF during RW pump refurbishment by considering the impact of fires on the RW pumps identified by the licensee, reviewing the IPEEE analysis, and conservatively assuming that the conditional core-damage probability (CCDP) during a fire that impacted an operating (not undergoing refurbishment) RW pump was equal to 0.1. Use of the 0.1 CCDP is reasonable since core cooling using the steam generators can be achieved using either the turbine-driven emergency feedwater pump, the diesel-driven emergency feedwater pump, or the main feedwater system, none of which depend on the RW system. The following table provides the results of this estimation process:

Fire Zone	Description	Fire-Induced Failure of the RW Pumps			Fire Freque ncy	Baseline CDF	CDF During RW Pump Refurbishment	
		1	2A	2B			2A	2B
CC-108-106	Battery Charger Room 3A	no	no	no	3.7E-4	1.5E-5	3.7E-5	3.7E-5
CC-108-108	4160V ES Switchgear Bus Room 3A	yes	yes	no	2.6E-4	7.3E-6	7.3E-6	2.6E-5
CC-108-107	4160V ES Switchgear Bus Room 3B	no	no	yes	2.3E-4	6.8E-6	2.3E-5	6.8E-6

Fire Zone	Description	Fire-Induced Failure of the RW Pumps			Fire Frequency	Baseline CDF	CDF During RW Pump Refurbishment	
		1	2A	2B			2A	2B
CC-124-117	480V ES Switchgear Bus Room 3A	no	yes	no	2.0E-4	3.8E-6	3.8E-6	2.0E-5
CC-108-105	Battery Charger Room 3B	no	yes	no	4.0E-4	2.7E-6	2.7E-6	4.0E-5
CC-108-102	Hallway and Remote Shutdown Room	no	yes	yes	1.2E-4	2.7E-6	2.7E-6	2.7E-6
CC-124-111	CRD and Communication Equipment Room	no	yes	yes	5.1E-4	1.6E-6	1.6E-6	1.6E-6
CC-108-109	Inverter Room 3B	no	no	no	2.1E-4	1.5E-6	2.1E-5	2.1E-5
CC-145-118B	Control Room	yes	yes	yes	1.2E-4	5.7E-7	5.7E-7	5.7E-7
CC-134-118A	Cable Spreading Room	yes	yes	yes	9.7E-5	9.9E-8	9.9E-8	9.9E-8
					TOTAL	4.2E-5	1.0E-4	1.6E-4

The IPEEE considered the impact of internal fires on containment performance. As stated in Chapter 4 of NUREG-1407, “For purposes of an IPEEE, a Level 1 probabilistic risk assessment (PRA) is considered acceptable to identify potential internal fire vulnerabilities at nuclear power plants.” Section 4.1.5 of NUREG-1407 instructed licensees to “Perform containment analysis if containment failure modes differ significantly from those found in the IPE internal events analysis.” The licensee’s IPEEE analysis concluded that since all of the fire zones with a CDF of greater than 1E-6/y were located in the control complex, none were adjacent to the containment. Therefore, the licensee has concluded that any fire which could possibly cause a containment integrity failure would not be located in a fire zone in which there was a significant potential for core damage due to the fire. The NRC staff review of the licensee’s IPE back-end analysis indicates that the major contributor to large early release arises from containment bypass scenarios primarily associated with steam generator tube ruptures (SGTRs). Based on these considerations, the NRC staff has determined that the contribution of internal fires to LERF is very small, and that the planned RW pump refurbishment activities do not measurably change this contribution.

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 5.0E-7 and the ICLERP is less than 5.0E-8. Using information provided by the licensee and including the impact of fire-related risks, the NRC staff estimated the following risk metrics:

RWP-2A REFURBISHMENT: ICCDP					
Configuration	CDF From Internal Initiators	CDF From Internal Fires	Total CDF	Change in CDF (ΔCDF)	Time In Configuration
Baseline	2.6E-6/y	4.2E-5/y	4.5E-5/y	n/a	n/a
RWP-2A and RWP-3A unavailable	6.5E-5/y	1.0E-4/y	1.7E-4/y	1.3E-4/y	6/365 = 0.0164 y

Only RWP-2A unavailable	2.1E-5/y	1.0E-4/y	1.2E-4/y	7.5E-5/y	(10-6)/365 = 0.0110 y
ICCDP (sum over all configurations of ΔCDF x time in configuration)				3E-6	

RWP-2A REFURBISHMENT: ICLERP					
Configuration	LERF From Internal Initiators	LERF From Internal Fires	Total LERF	Change in LERF (ΔLERF)	Time In Configuration
Baseline	3.59E-7/y	very small	3.59E-7/y	n/a	n/a
RWP-2A and RWP-3A unavailable	4.06E-7/y	very small	4.06E-7/y	4.70E-8/y	6/365 = 0.0164 y
Only RWP-2A unavailable	3.82E-7/y	very small	3.82E-7/y	2.30E-8/y	(10-6)/365 = 0.0110 y
ICLERP (sum over all configurations of ΔLERF x time in configuration)				1.0E-9	

RWP-2B REFURBISHMENT: ICCDP					
Configuration	CDF From Internal Initiators	CDF From Internal Fires	Total CDF	Change in CDF (ΔCDF)	Time In Configuration
Baseline	2.6E-6/y	4.2E-5/y	4.5E-5/y	n/a	n/a
RWP-2B unavailable	2.7E-6/y	1.6E-4/y	1.6E-4/y	1.2E-4	10/365 = 0.0274 y
ICCDP (sum over all configurations of ΔCDF x time in configuration)				3E-6	

The licensee did not provide an estimate of the change in LERF during RWP-2B replacement. Based on the very small ICLERP associated with RWP-2A refurbishment, the NRC staff concludes that the ICLERP associated with RWP-2B refurbishment would be very small.

The ICCDP values associated with RWP-2A and RWP-2B replacement are above the RG 1.177 risk acceptance guidelines. The NRC staff concludes that the risk impact of the proposed one-time increase in the CT for restoring an inoperable nuclear services seawater system train to operable status is small and acceptable for the following reasons:

- The proposed license amendment concerns a one-time (temporary) change to the TS. As previously noted, RG 1.177 is directly applicable only to permanent changes to TS requirements. Further, the risk acceptance guidelines RG 1.177 should not applied in an overly prescriptive manner

- The NRC staff'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.

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. The "delta risk" metrics (ΔCDF and ΔLERF) of RG 1.174 reflect the change in risk due to the increase in annualized RW pump unavailability associated with the one-time RW pump refurbishment activities. Using information provided by the licensee and including the impact of fire-related risks, the NRC staff estimated the following risk metrics:

RWP-2A REFURBISHMENT: CORE-DAMAGE FREQUENCY				
Configuration	CDF From Internal Initiators	CDF From Internal Fires	Total CDF	Fraction of Time In Configuration
Baseline	2.6E-6/y	4.2E-5/y	4.5E-5/y	(8760-120)/8760 hours = 0.9863
RWP-2A and RWP-3A unavailable	6.5E-5/y	1.0E-4/y	1.7E-4/y	60/8760 hours = 0.0068
Only RWP-2A unavailable	2.1E-5/y	1.0E-4/y	1.2E-4/y	(120-60)/8760 hours = 0.0068
CDF considering RWP-2A refurbishment (weighted average)			4.6E-5/y	
Change in CDF (ΔCDF)			1E-6/y	

RWP-2A REFURBISHMENT: LARGE EARLY RELEASE FREQUENCY				
Configuration	LERF From Internal Initiators	LERF From Internal Fires	Total LERF	Fraction of Time In Configuration
Baseline	3.59E-7	very small	3.59E-7/y	(8760-120)/8760 hours = 0.9863
RWP-2A and RWP-3A unavailable	4.06E-7	very small	4.06E-7/y	60/8760 hours = 0.0068
Only RWP-2A unavailable	3.82E-7	very small	3.82E-7/y	(120-60)/8760 hours = 0.0068
LERF considering RWP-2A refurbishment (weighted average)			3.59E-7/y	
Change in LERF (ΔLERF)			5E-10/y	

RWP-2B REFURBISHMENT: CORE-DAMAGE FREQUENCY				
Configuration	CDF From Internal Initiators	CDF From Fires	Total CDF	Fraction of Time In Configuration
Baseline	2.6E-6/y	4.2E-5/y	4.5E-5/y	(8760-72)/8760 hours = 0.9918
RWP-2B unavailable	2.7E-6/y	1.6E-4/y	1.6E-4/y	72/8760 hours = 0.0082
CDF considering RWP-2A refurbishment (weighted average)			4.6E-5/y	
change in CDF (ΔCDF)			1E-6/y	

The licensee did not provide an estimate of the change in LERF during RWP-2B replacement. Based on the very small change in LERF associated with RWP-2A refurbishment, the NRC staff concludes that the change in LERF associated with RWP-2B refurbishment would be very small.

Considering the information presented above, including the conservatisms and uncertainties in the analysis, the NRC staff concludes that the risk impact of the proposed one-time increase in the CT for restoring an inoperable nuclear services seawater system train to operable status lies in Region III of Figures 3 and 4 contained in RG 1.174. Therefore, in accordance with the RG 1.174 risk acceptance guidelines, the licensee's proposed license amendment to allow a one-time extension of the RW train CT from 72 hours to 10 days 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.

3.4.2.2 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, is 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 RW pump refurbishment activities, the license reviewed its PRA results to identify compensatory measures that minimize risk. These compensatory measures include:

- The 4160-A Engineered Safeguards (ES) Bus is normally aligned to the OPT and the 4160-B ES Bus is aligned to the BEST. This compensatory measure considers changing the alignment of the 4160-B ES Bus to the OPT to keep RWP-2B continuously powered and available for autostart in case of a partial loss of offsite power event. A loss of offsite power to the OPT would not cause a plant trip. The 4160-A ES Bus will be aligned to the BEST for redundancy. This compensatory measure has a more significant benefit effect on lowering CDF for RWP-2A refurbishment activity due to plant asymmetries.

- Selection of beneficial makeup pump configurations.

- Increase of operator attention to loss/restoration of RW/SW redundant train control complex chiller (e.g., walkdowns of the operable redundant train, pre-job discussion on the impact of losing service water and the potential emergency feedwater initiation and control (EFIC) system control problems if there is a loss of the redundant train control complex chiller.

- Operator attention to potential use of the Appendix R chiller, non-safety-grade FWP-7 and standby diesel generator (MTDG-1).

- Periodic operator walkdowns of the redundant train.

- No electric maintenance to be scheduled in the switchyard that would challenge the availability of offsite power to the ES Buses or to the bus for RWP-1.

- Establishing a periodic fire watch in fire zones identified as containing circuits applicable to the RW/SW pumps to minimize fire risk in these areas. Those fire zones were identified by a review of the CR-3 IPEEE.

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.

3.4.2.3 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 one-time extended CT, License Condition 2.C.(13) requires certain compensatory measures to be implemented during the RW pump 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 CR-3 EOOS computer model provides the licensee with 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.

3.5 NRC Staff Findings

In summary, the NRC staff finds that the licensee's proposed change to revise, on a one-time basis, the CT for ITS 3.7.9, Condition A, Required Action A.1 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. Thus, the NRC staff has concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in this 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.

4.0 REGULATORY COMMITMENTS

Subsequent to the licensee's original submittal, the licensee indicated that the proposed one-time CT extension would be only utilized to refurbish RWP-2B since repair work on RWP-2A had been completed. The licensee has provided the following license conditions that apply during refurbishment of nuclear services seawater system emergency pump RWP-2B:

1. No elective maintenance will be scheduled on other related risk sensitive equipment beyond that required for the refurbishment activity of RWP-2B that could degrade the risk profile of the plant. For these purposes, the systems are: Nuclear Services and Decay Heat Seawater System, Decay Heat System, Decay Heat Closed Cycle Cooling Water System, Nuclear Services Closed Cycle Cooling Water (SW), Emergency Diesel Generators, Chilled Water, Emergency Feedwater System, Emergency Feedwater Initiation and Controls System (EFIC) Auxiliary Feedwater Pump and Makeup System.

2. Makeup Pump configurations will be selected to minimize risk.

3. Increased operator attention will be focused on loss/restoration of RW/SW/ redundant train and the Appendix R Chiller. This will be accomplished by on shift operating crew review of Abnormal Procedure (AP)-330, Loss of Nuclear Service Cooling.

4. Operator attention will be focused on potential use of non-safety grade Feedwater Pump FWP-7 and its dedicated Diesel Generator (MTDG-1). This will be accomplished by on shift operating crew review of Emergency Operating Procedure (EOP)-14, Enclosure 7, Emergency Feedwater Pump (EFWP) Management.

5. Daily operator walkdowns of the redundant train of RW/SW/Pumps and associated power supply switchgear will be conducted.

6. No elective maintenance to be scheduled in the switchyard that would challenge the availability of offsite power to the ES Buses or to the Bus for RWP-1.

7. Hourly roving fire watches will be established in fire zones identified as containing circuits required for RWP-1 or RWP-2A to minimize fire risk in these areas. Those fire zones are: AB-95-3AA, AB-95-3B, AB-95-3E, AB-95-3F, AB-95-3G, AB-95-3K, AB-95-3T, AB-95-3U, AB-95-3W, AB-95-3X, AB-95-3Z, CC-108-102, CC-108-105, CC-108-105, CC-108-106, CC-108-108, CC-108-109, CC-108-110, CC-124-111, CC-124-116, CC-124-117, CC-134-118A, IB-95-200C, IB-119-201B, TB-119-400E, TB-119-403, TB-95-400A, TB-95-401, AB-119-6, and AB-119-6A.

8. Prior to entering the Allowed Outage Time for rebuild of RWP-2B, fire zones containing circuits for either RWP-1 or RWP-2A will be walked down to identify and minimize transient combustibles not related to ongoing approved work. Hourly roving fire watches will be charged with continuing to monitor for the presence of transient combustible materials until RWP-2B is returned to service.

9. For the risk significant fire zones containing circuits for both RWP-1 and RWP-2A, the following additional compensatory measures will be established which address the specific risk factors in each zone. No additional compensatory measures beyond roving fire watches will be established for fire zones that contain circuits for all three pumps (RWP-1, RWP-2A and RWP-2B).

Fire Zone CC-108-108
A continuous Fire Brigade Qualified fire watch will be stationed in the fire area except while 4160V breaker manipulations are being performed. The individual's turnout gear will be available in the adjacent area where a fire hose station is also located equipped with an electrically safe fog nozzle.

Fire Zone AB-95-3W
The Waste Transfer Pumps will only be operated when there is a qualified fire watch in the immediate vicinity of the operating pump equipped with a radio and fire extinguisher. No hot or spark producing work will be conducted. A roving hourly fire watch will observe this zone.

Fire Zone AB-95-3E
The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.

Fire Zone AB-95-3AA
The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.

Fire Zone AB-95-3F
The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.

Fire Zone AB-95-3T
The portion of the fire zone that can be locked will be locked and the keys will be administratively controlled. Entries will be limited to only operationally necessary activities and require inspection for transient combustible materials upon exit. The portion of the zone that cannot be locked will be observed by the roving fire watch. No hot or spark producing work will be conducted.

Fire Zone AB-95-3U
The portion of the fire zone that can be locked will be locked and the keys will be administratively controlled. Entries will be limited to only operationally necessary activities and require inspection for transient combustible materials upon exit. The portion of the zone that cannot be locked will be observed by the roving fire watch. No hot or spark producing work will be conducted.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments 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 SUMMARY

The NRC staff finds that the licensee's proposed change to revise, on a one-time basis, the CT for ITS 3.7.9, Condition A, Required Action A.1 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.

6.0 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.

7.0 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 (68 FR 46244). 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.

8.0 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.

9.0 REFERENCES

1. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, “Crystal River Unit 3 - License Amendment Request #280, Revision 0, Revised Improved Technical Specification (ITS) 3.7.9, Nuclear Services Seawater System,” 3F0703-04, July 14, 2003.

2. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, “Crystal River Unit 3 - Response to Request for Additional Information Regarding License Amendment Request #280, Revision 0, Revised Improved Technical Specification 3.7.9, Nuclear Services Seawater System,” 3F1103-06, November 20, 2003.

3. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, “Crystal River Unit 3 - Proposed License Condition for License Amendment Request #280, Revision 0, Revised Improved Technical Specification (ITS) 3.7.9, Nuclear Services Seawater System,” 3F0304-09, March 25, 2004.

4. Letter from Dale E. Young, Progress Energy Florida, Inc. to U.S. Nuclear Regulatory Commission, “Crystal River Unit 3 - Modified Proposed License Condition, License Amendment Request #280, Revision 1, Revised Improved Technical Specification (ITS) 3.7.9, Nuclear Services Seawater System,” 3F0404-11, April 27, 2004.

5. 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.

6. 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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