Progress Energy

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

Ref: 10 CFR 50.90

January 13, 2005
3F0105-02

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

Subject: 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

Dear Sir:

Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF), hereby submits License Amendment Request (LAR) #289, Revision 0, which requests a one-time change to the Crystal River Unit 3 (CR-3) Facility Operating License in accordance with 10 CFR 50.90. LAR #289 proposes a change to 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 System (DC) and 3.7.10, Decay Heat Seawater System. The changes will allow an increased allowed outage time (AOT) for up to 10 days to support the refurbishment of Decay Heat Seawater System Pump RWP-3B. The refurbishment activity is planned to occur at power operation, during the best available schedule opportunity.

The acceptability of the changes proposed in this submittal is supported by risk-informed considerations. This information is provided in Attachments A and E of this submittal.

A list of regulatory commitments is included in Attachment F. CR-3 will implement the provisions described in these commitments during the proposed one-time extended AOT. The provisions described in these commitments provide compensatory measures that will reduce or mitigate risks associated with having RWP-3B out-of-service for greater than 72 hours. The compensatory measures are based on risk insights and consideration of external events such as adverse weather conditions and fire. CR-3 will submit supplemental information regarding the risk significant fire zones including any additional specific compensatory measures by February 11, 2005.

PEF respectfully requests that this request be noticed in the Federal Register as soon as practical since it may be needed in an expedited basis.

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

The CR-3 Plant Nuclear Safety Committee has reviewed this request and recommended it for approval.

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

Sincerely,

Dale E Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/lvc

Attachments:

A. Background, Description of Proposed Change, Reason for Request, Evaluation of Request, Consideration of External Events, Risk Evaluation, Compensatory Measures, Performance Monitoring, Conclusion and Precedent
B. Regulatory Analysis (No Significant Hazards Consideration Determination, Applicable Regulatory Requirements and Environmental Impact Evaluation)
C. Proposed Revised Improved Technical Specifications Pages – Strikeout/Shadowed Format
D. Proposed Revised Improved Technical Specifications Pages – Revision Bar Format
E. PSA Risk Assessment of RWP-3B Extended AOT
F List of Regulatory Commitments

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

STATE OF FLORIDA

COUNTY OF CITRUS

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

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 13th day of January, 2005, by Dale E. Young.

Janet Schroeder

Signature of Notary Public
State of Florida

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

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

Personally Known ✓ -OR- Produced Identification ______

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #289, REVISION 0

Background, Description of Proposed Change, Reason for Request, Evaluation of Request, Consideration of External Events, Risk Evaluation, Compensatory Measures, Performance Monitoring, Conclusion and Precedent

Background

The Crystal River Unit 3 (CR-3) Decay Heat Seawater System contains two decay heat seawater pumps (RWP-3A and RWP-3B). RWP-3A takes suction from the "A" Raw Water Pit; RWP-3B takes suction from the "B" Raw Water Pit. The pits are supplied with water from the Gulf of Mexico. As explained in subsequent sections of this submittal, the system provides cooling water to the tube side of the two heat exchangers removing heat from the Decay Heat Closed Cycle Cooling Water (DC) System and subsequently rejects it to the ultimate heat sink (the Gulf of Mexico) through the discharge canal.

A recently performed operability assessment of Decay Heat Seawater pump RWP-3B demonstrated that although the pump remains operable, it exhibits a degraded flush flow condition. A refurbishment activity to restore the flush water flow to full qualification is being planned to occur at power operation during the best available schedule opportunity.

Improved Technical Specification (ITS) 3.7.10, "Decay Heat Seawater System," requires that two Decay Heat Seawater System trains shall be OPERABLE. If one train is inoperable, Condition "A" allows operation to continue for 72 hours. It is estimated that the rebuild activity of RWP-3B will take approximately 5 days. Thus, to perform the refurbishment activity online, a one-time allowed outage time (AOT) extension of the ITS 3.7.10 Completion Time to 10 days is needed. Other systems affected by the extended AOT needed to refurbish RWP-3B require their AOT to also be extended to 10 days. However, no maintenance is being performed on those systems.

Description of Proposed Change

License Amendment Request (LAR) #289, Revision 0, is proposing a one-time extended AOT change to 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 System (DC) and 3.7.10, Decay Heat Seawater System to allow the refurbishment of Decay Heat Seawater System Pump RWP-3B online.

ITS 3.5.2, ECCS – Operating

The following footnote will be added to allow for one-time only, one train of ECCS – Operating to be inoperable for up to 10 days.

> **"*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 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."**

ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems

The following footnote will be added for the Reactor Building Spray and Containment Cooling Systems to allow for one-time only, one train of Reactor Building Spray and Containment Cooling Systems to be inoperable for up to 10 days.

> **"*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 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."**

ITS 3.7.8, DC System

The following footnote will be added for the DC System to allow for one-time only, one train of DC System to be inoperable for up to 10 days.

> **"*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 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."**

ITS 3.7.10, Decay Heat Seawater System

The following footnote will be added for the Decay Heat Seawater System to allow for one-time only, one train of Decay Heat Seawater System to be inoperable for up to 10 days.

> **"*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."**

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

"*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."

Reason for Request

Since the duration of the repair activity to return RWP-3B to full qualification is greater than the 72 hour Completion Time specified in ITS 3.7.10, the repair can not be performed in MODES 1 to 4 unless a one-time extension of the Completion Time to 10 days is approved. Other systems affected by the extended AOT needed to refurbish RWP-3B also require their AOT to be extended to 10 days. Approval of the proposed LAR will allow the performance of the maintenance activity during power operations.

Evaluation of Request

System Description

Emergency Core Cooling System (ECCS)

The ECCS provides core cooling to ensure the reactor core is protected after a Design Basis Accident. The ECCS consists of two redundant, 100% capacity trains. Each train consists of high pressure injection (HPI) and low pressure injection (LPI) subsystems. As reflected in the attached risk calculation (Attachment E), the low pressure injection and high pressure recirculation function of "B" train will be unavailable.

During the time that RWP-3B is out of service, the operable ECCS train will be protected by minimizing maintenance on the system. In this configuration, the operable train will be capable of responding as designed during design basis events. Therefore, only the redundancy of the ECCS is affected by the extension of the required action.

The requested one-time extension from 72 hours to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online is reasonable considering the redundant capabilities of the system, the proposed compensatory actions that will be taken and the risk considerations as discussed later in this attachment.

Reactor Building Spray and Containment Cooling Systems

The Reactor Building (RB) Spray and Containment Cooling Systems limit the temperature and pressure that could be experienced following a design basis accident. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment. The systems consist of two separate trains of equal capacity and each train is capable of performing the iodine removal and containment cooling functions. As reflected in the attached risk calculation (Attachment E), only the RB spray "B" train will be unavailable.

During the time that RWP-3B is out of service, the operable RB Spray and Containment Cooling Systems train will be protected by minimizing maintenance on the system. In this configuration, the operable train will be capable of responding as designed during design basis events. Therefore, only the redundancy of the RB Spray and Containment Cooling System is affected by the extension of the required action.

The requested one time extension from 72 hours to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online is reasonable considering the redundant capabilities of the system, the proposed compensatory actions that will be taken and the risk considerations as discussed later in this attachment.

Decay Heat Closed Cycle Cooling Water System (DC)

The DC System facilitates the removal of decay heat from the reactor core. The system also removes process and operating heat from safety related components associated with decay heat removal during plant cooldown and following a transient or accident. The DC System is not a normally operating system, but must be capable of performing its post-accident safety functions, which include providing cooling water to components required for Reactor Building Spray System and containment heat removal. One DC train is adequate to perform the heat removal function.

During the time that RWP-3B is out of service, the operable DC System train will be protected by minimizing maintenance on the system. In this configuration, the operable train will be capable of responding as designed during design basis events. Therefore, only the redundancy of the DC System is affected by the extension of the required action.

The requested one-time extension from 72 hours to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online is reasonable considering the redundant capabilities of the system, the proposed compensatory actions that will be taken and the risk considerations as discussed later in this attachment.

Decay Heat Seawater System and Nuclear Services Seawater System (RW)

The Decay Heat Seawater System and the Nuclear Services Seawater System comprise the RW system which is shown in simplified schematics in Figure 1. Seawater is drawn from the intake canal and conveyed to the sump pit via two redundant 48-inch intake conduits. The "A" intake conduit shares a common intake structure, bar racks, and traveling screens with the Circulating Water System (CW) system while the other 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: One 100% capacity normal nuclear services seawater pump, one 100% capacity emergency nuclear services seawater pump, and one 100% capacity decay heat service seawater pump in the "B" compartment; and another group of one 100% capacity emergency nuclear services seawater pump and one 100% capacity decay heat service seawater pump in the "A" compartment. The Decay Heat Seawater pumps (Figure 1) supply flow to the Decay Heat Closed Cycle Cooling Water (DC) System heat exchangers which facilitates the removal of decay heat from the reactor core during design basis accidents and normal plant shutdowns. Each Decay Heat Seawater pump is powered from a separate 4160 volt Engineered Safeguards (ES) Bus.

The Decay Heat Seawater pumps are designed to the parameters shown below:

Decay Heat Seawater Pumps	
Number	2
Flow, gpm	9,700
Design Head, ft	75
Design Pressure, psig	75
Design Temperature, °F	109
Seismic Class	I

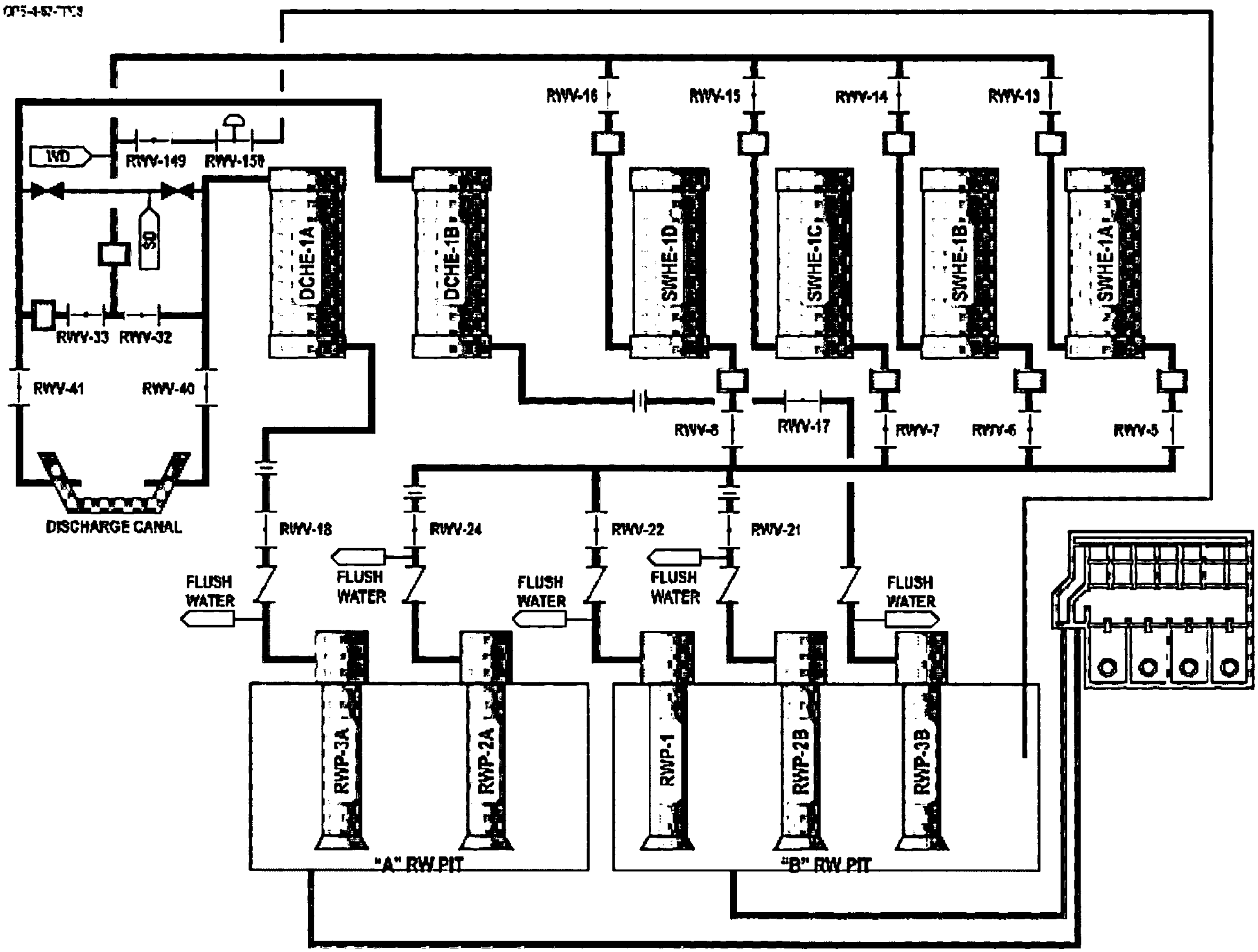

Figure 1 - RW System Flowpaths

Technical Evaluation

During the requested extended time period of 10 days, the redundant Decay Heat Seawater pump RWP-3A will be available and capable of providing cooling for decay heat removal loads and essential equipment during emergency conditions.

To ensure defense-in-depth capabilities, and that the assumptions of the risk assessment are maintained during the proposed one-time extended Completion Time, CR-3 will continue the performance of 10 CFR 50.65(a)(4) assessments before performing maintenance or surveillance activities and no maintenance activities of other risk sensitive equipment beyond that required for the refurbishment activity will be scheduled concurrent with the repair activity. Other compensatory actions that will be implemented include: operator attention to the importance of protecting the operable redundant train will be increased, selection of beneficial Makeup Pump configurations, no elective maintenance to be scheduled in the switchyard, and the establishment of fire watches (the information regarding risk significant fire zones and appropriate compensatory measures will supplement this submittal by February 11, 2005). These actions are more fully described in subsequent sections.

Consideration of External Events

The external events assessed for this activity include severe weather phenomena and internal fires. These events were evaluated as explained below and compensatory measures were determined to minimize risk during the planned RWP-3B maintenance.

Severe Weather Phenomena

CR-3 utilizes Emergency Preparedness procedure EM-220 to address violent weather. This procedure provides guidance during the onset of a Flood Warning, Tornado Watch, Tornado Warning, Tropical Storm Watch, Tropical Storm Warning, Hurricane Watch, and Hurricane Warning; and implements activities to ensure plant safety by having the necessary equipment and supplies on hand. This procedure also provides guidance in recovering from the effects of a hurricane and implements Institute of Nuclear Plant Operations (INPO) Significant Operating Event Report (SOER) 02-01, Severe Weather. EM-220 defines conditions when operators must place the plant in a shutdown condition for violent weather conditions.

CR-3 is designed to maintain safe shutdown capability in the event of a postulated hurricane. The following equipment is required to remain functional during the postulated hurricane to assure maintenance of the reactor in a safe condition:

a. Onsite Emergency Diesel Generators, and their support equipment (fuel systems, cooling systems, switchgear)

b. Reactor Decay Heat Removal equipment:

    1. Nuclear Services Closed Cycle Cooling System (Service Water) (SW)
    2. Decay Heat Removal System (DH)
    3. Decay Heat Closed Cycle Cooling System (DC)
    4. Nuclear Services and Decay Heat Seawater System (Raw Water) (RW)

Ability of this equipment to remain functional is assured by the facility design, as follows: Onsite power generation equipment is located within the auxiliary building, which is protected from flooding by water-tight doors at all openings lower than the predicted wave run-up height. Fuel storage tanks are located underground and are restrained against damage from their own buoyancy by hold-down straps and concrete anchor slabs. Tank vents are above postulated wave tops to prevent seawater entering the tanks via the vent lines. Diesel engine cooling is accomplished by a self-contained air radiator system within the structure. The SW System, DH System, and DC System are all located within the auxiliary building and are powered by onsite diesel generators. Additional component protective facilities required for local protection are discussed in the CR-3 Final Safety Analysis Report, Chapter 2.4.2, Flood Studies and Hurricane Effects.

Internal Fire

The risk insights for fires are primarily based on a combination of portions of the Individual Plant Evaluation of External Events (IPEEE) and Safe Shutdown studies for CR-3.

Because the plant configuration has changed since the IPEEE was first developed, a quantitative fire risk assessment was not performed, and a more qualitative defense in depth approach was used. Evaluation of equipment that would be available in the event of a fire is based on equipment evaluations and circuit locations within fire areas using the deterministic criteria of Appendix R. Those evaluations postulate that if equipment and circuits are not separated or protected in accordance with Appendix R III.G.2 criteria, then they will be made unavailable due to a fire in that area. The risk-informed portion of this evaluation, however, does consider the potential that equipment, which provides defense in depth but does not meet Appendix R criteria, may be available.

Risk Evaluation

Attachment E provides the calculation performed to assess the risk associated with increasing the ITS Completion Time (one-time from 72 hours to 10 days) to perform repairs to RWP-3B. The calculation includes the risk associated with having a Decay Heat Service Seawater pump out-of-service for 10 days using the current CR-3 Equipment Out-Of-Service (EOOS) computer model based on the most current plant Probabilistic Safety Analysis (PSA).

Current licensing conditions at CR-3 require transition to mode 5 if RWP-3B is inoperable for more than 72 hours. Because the RW-DC system is vital for shutdown cooling there is also some increase in risk if this activity were performed while shutdown. Performing this activity with a "hot" Reactor Coolant System (RCS) provides more options for the use of Steam Generator cooling for RCS heat removal. CR-3s current PSA does not quantify the specific risk involved, however, it is clear that the current assessment is conservative in this respect.

The Technical Specification Task Force Change Traveler, TSTF-430,R.2, "AOT Extension to 7 Days for LPI and Containment Spray," was approved by the NRC on a Safety Evaluation (SE) dated August 5, 2004. The NRC stated in the SE: "The proposed change is expected to have a beneficial impact on the risk during the shutdown period, no impact during the transition period, and a small impact on the risk during power operation when the compensatory measures are implemented." The NRC also stated in the Conclusion section of the SE: "Additionally, the requested AOT extensions to 7 days for the Decay Heat Closed Cycle Cooling and Decay Heat Seawater systems for Crystal River, Unit 3 are supportable at this time."

The PSA risk associated with the activity to repair the Emergency Nuclear Services Seawater pump supports the one-time extension proposed in this LAR. Assuming RWP-3B is out-of-service for 10 days, the bounding risk due to internal events for this activity is estimated with a Change in Core Damage Frequency (ΔCDF) of 1.5E-06 and a sensitivity based on fire risk add about 2.72E-06. The total risk is below the Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," limit of 1E-05 and is considered a small increase. Compensatory actions to reduce the probability of fire and to enhance fire detection and suppression in the more vulnerable areas can be used to reduce the risk. The Change in Large Early Release Frequency (ΔLERF) for all cases evaluated is well below the RG 1.174 limit of 1E-07 and is considered very small. The risk evaluation concludes that the one-time 10-day Completion Time proposed in this LAR results in a ΔCDF and a ΔLERF that is reasonable compared to the criteria in RG 1.174. Although not directly applicable, those results are also reasonable when compared to the guidance in RG 1.177, "An Approach for Plant-Specific Risk-Informed Decisionmaking: Technical Specifications."

Quality of the Crystal River Unit 3 PSA

The models used for this application were generated using updated Individual Plant Examination (IPE) models developed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and associated supplements. The original development work was a level one Probabilistic Risk Assessment (PRA) study completed in 1987 (Crystal River Unit 3 Probabilistic Risk Assessment, Florida Power Corporation, Science Applications Intl. Corporation, July 1987), which was submitted to the NRC and reviewed by Argonne National Laboratory (NUREG/CR-5245). This study was subsequently updated for the Generic Letter 88-20 IPE submittal to include a level two containment analysis and an internal flooding analysis.

Revisions to the models have been made to maintain the models consistent with plant design changes and operational changes. These changes have been made by individuals knowledgeable in risk assessment techniques and methods, and reviewed by plant Engineering and Operations personnel familiar with the plant design and operation. The current PSA model and the risk assessment performed for this application have been documented as a calculation.

Current administrative controls include written procedures and review of all model changes, data updates, and risk assessments performed using PSA methods and models. Risk assessments are performed by a PSA engineer, reviewed by another PSA engineer, and approved by the PSA Supervisor or designee. Procedures, PSA model documentation, and associated records for applications of the PSA models are controlled documents.

Since the submittal of the original PRA study in 1987, the PSA models have been maintained consistent with the current plant configuration such that they are considered "living" models which reflect the as-built, as-operated plant. The PSA models are updated for different reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, and advances in PSA technology. The update process ensures that the applicable changes are implemented and documented in a timely manner so that risk analyses performed in support of plant operations reflect the current plant configuration, operating philosophy, and transient and component failure history. The PSA maintenance and update process is described in administrative procedure ADM-NGGC-0004, "Updates to PSA Models." Model updates are performed at a frequency dependent on the estimated impact of the accumulated changes. Guidance to determine the need for a model update is provided in the procedure. Prior to startup from a refueling outage, known outstanding changes, including identified model errors and enhancements, are reviewed, and either model changes are implemented, or the outstanding item is dispositioned to be deferred for a future model update.

PSA Software

Computer programs that process PSA model inputs are verified and validated in accordance with administrative procedure CSP-NGGC-2505, "Software Quality Assurance and Configuration Control of Business Computer Systems." This procedure provides for software verification and validation to ensure the software meets the software requirement specifications and functional requirements, and typically includes a comparison of results generated to the results generated from previously approved software.

Validation requirements for each quality related PSA computer program are documented in the Software Life Cycle document, which consists of a Software Verification/Validation Plan (SVVP) and Report (SVVR). These requirements include the method of validation, the frequency of validation, the documentation required and the acceptance criteria. Actual validation benchmark problems can exercise more than one program, but a separate SVVR must be submitted for each program. Each SVVP and SVVR is reviewed, and then approved by the software owner, who is the PSA Supervisor. Software validation tests both the software and the hardware. Validation tests are also performed following any significant change in the hardware, operating system, or program, or if the validation period established in the SVVP procedure expires.

Model Changes Since Submittal of the IPE

Since the submittal of the IPE, there has been several significant plant design changes incorporated into the PSA model which have resulted in a reduction in the Core Damage Frequency (CDF). Updates have been made to plant-specific data (through 1999) and initiating events data, as well as updates to the methods used for human reliability, common cause, internal flooding and level two analyses.

As of the date of this submittal, there are no outstanding or planned plant changes requiring a change to the PSA model which would affect the conclusions of the analysis in Attachment E.

PSA Reviews

As discussed above, the original CR-3 PRA study was reviewed by Argonne National Laboratory as documented in NUREG/CR-5245. For the IPE submittal, multiple levels of review were used, including an assessment by Engineering and Operations personnel familiar with the plant design and operation. Subsequent revisions to the PSA models were performed by qualified individuals with knowledge of PSA methods and plant systems. Involvement by Engineering and Operations personnel in providing input and review of results was obtained, when required, based on the scope of the changes being implemented.

The CR-3 PSA 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 failure analyses were updated to current industry methodologies and data sources. An internal review of the PSA model elements and their corresponding documentation was conducted to assure the model and documentation reflected the plant design.

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

Following completion of this review, the CR-3 PSA model was revised to address each issue identified which affected the model. The significant changes identified included:

- Update of plant-specific thermal-hydraulic analyses which provide the bases for accident sequences, system success criteria, and timing for operator actions
- Revision of accident sequence logic for steam generator tube rupture (SGTR) and anticipated transient without scram (ATWS) mitigation
- Development of an initiating event to address the loss of all raw water pumps (loss of ultimate heat sink)
- Update of the interfacing systems loss of coolant accident (ISLOCA) analyses
- Update of the human reliability analysis including the dependency analysis for multiple operator action responses to an event, and
- Update of the level two analysis

Issues involving model documentation are being addressed as each individual PSA 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 PSA models (including the Robinson Nuclear Plant, Brunswick Nuclear Plant, and Harris Nuclear Plant). The issues identified by the peer review in these areas have been reviewed and determined not to have any impact on this submittal, and so deferral of completion of these items is acceptable for this application of the PSA model. All other peer review items which impact the PSA model have been addressed and are reflected in this submittal.

At the time of the peer review, the level two model was not yet completed, and only a preliminary draft version, along with the original IPE level two results, were available for review. The level two model is now complete, and the findings identified from the peer certification review of the preliminary results and the IPE model have been addressed.

Compensatory Measures

The PSA Risk Assessment assumes the continued performance of 10 CFR 50.65(a)(4) assessments before performing maintenance or surveillance activities. It also assumes that no maintenance will be scheduled on other related risk sensitive equipment beyond that required for the 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), Auxiliary Feedwater Pump].

Although the risk associated with the proposed maintenance activity is considered small without taking special actions, the compensatory actions listed below can further reduce the risk:

1. 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.

2. CR-3 will select beneficial Makeup Pump configurations.

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

4. 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 Operating Procedure (EOP-14), Enclosure 7, Emergency Feedwater Pump (EFWP) Management.

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

6. 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.

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

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

Performance Monitoring

All equipment relied upon for supplying electric power and mitigating loss of power events is included in the CR-3 Maintenance Rule Program and is monitored for equipment unavailability.

Conclusion

CR-3 has evaluated the risks associated with an extended AOT to refurbish RWP-3B. CR-3 has also identified the fire and adverse weather events as significant concerns. In order to reduce the risk from these events, CR-3 has proposed compensatory measures that maximize the availability

availability of power sources, increase operator awareness and avoid risk due to violent weather through preparation and planning. PEF believes that approval of the proposed change to ITS 3.5.2, ECCS, 3.6.6, RB Spray and Containment Cooling Systems, 3.7.8, DC System and 3.7.10, Decay Heat Seawater System will pose an insignificant increase in risk to the plant or to the health and safety of the public.

Precedent

There are similarities between this LAR and a LAR that NRC approved for CR-3 regarding the Nuclear Services Seawater System in License Amendment No. 212, issued on May 18, 2004 (TAC No. MC0110). There are also similarities to License Amendments Nos. 203 and 196 granted for Catawba Nuclear Station on January 7, 2003.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #289, Revision 0

Regulatory Analysis (No Significant Hazards Consideration Determination, Applicable Regulatory Requirements and Environmental Impact Evaluation)

No Significant Hazards Consideration Determination

License Amendment Request (LAR) #289, Revision 0, proposed changes include a one-time change to Improved Technical Specifications (ITS) for the systems affected during the refurbishment of Decay Heat Seawater System Pump RWP-3B. This will allow refurbishment of RWP-3B pump during power operations.

This LAR proposes to extend the Completion Time, Required Action A.1 from 72 hours to 10 days of 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 to allow the refurbishment of RWP-3B. This request has been evaluated against the standards in 10 CFR 50.92, and has been determined to not involve a significant hazards consideration. In support of this conclusion, the following analysis is provided:

1. *Does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed license amendment extends, on a one-time basis, the Completion Time for the systems described above from 72 hours to 10 days. These Systems are designed to provide cooling for components essential to the mitigation of plant transients and accidents. The systems are not initiators of design basis accidents. The proposed ITS changes have been evaluated to assess their impact on normal operation of the systems affected and to ensure that their design basis safety functions are preserved.

A Probabilistic Safety Assessment (PSA) has been performed to assess the risk impact of an increase in Completion Time from 72 hours to 10 days. Although the proposed one-time change results in an increase in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), the value of these increases are considered as small (CDF) and very small (LERF) in the current regulatory guidance.

Therefore, granting this LAR does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does not create the possibility of a new or different type of accident from any accident previously evaluated.*

The proposed license amendment extends, on a one-time basis, the Completion Time for the systems described above from 72 hours to 10 days.

The proposed LAR will not result in changes to the design, physical configuration of the plant or the assumptions made in the safety analysis. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. *Does not involve a significant reduction in the margin of safety.*

The proposed license amendment extends, on a one-time basis, the Completion Time for the systems described above from 72 hours to 10 days. The proposed change will allow online repair of Decay Heat Seawater pump RWP-3B to restore the pump to full qualification which will improve its reliability and useful lifetime, thus increasing the long term margin of safety of the system.

The proposed LAR will reduce the probability (and associated risk) of a plant shutdown to repair a Decay Heat Services Seawater pump. To ensure defense-in-depth capabilities and the assumptions in the risk assessment are maintained during the proposed one-time extended Completion Time, CR-3 will continue the performance of 10 CFR 50.65(a)(4) assessments before performing maintenance or surveillance activities and no maintenance activities of other risk sensitive equipment beyond that required for the refurbishment activity will be scheduled concurrent with the repair activity. Other compensatory actions that will be implemented include: operator attention to the importance of protecting the operable redundant train and support systems will be increased, selection of beneficial Makeup Pump configurations, no elective maintenance will be scheduled in the switchyard, and the establishment of fire watches.

As described above in Item 1, a PSA has been performed to assess the risk impact of an increase in Completion Time. Although the proposed one-time change results in an increase in Core Damage Frequency (CDF), and Large Early Release Frequency (LERF), the value of these increases are considered as small (CDF) and very small (LERF) in the current regulatory guidance.

Therefore, granting this LAR does not involve a significant reduction in the margin of safety.

Based on the above, Progress Energy Florida, Inc. (PEF) concludes that the proposed LAR presents a no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

Applicable Regulatory Requirements

PEF has evaluated the Regulatory Requirements applicable to the proposed changes to the ITS for the systems affected during the refurbishment of Decay Heat Seawater System Pump RWP-3B. These requirements include 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. PEF has determined that the proposed change does not require any exemptions or relief from regulatory requirements other that the changes requested to.

Environmental Impact Evaluation

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) result in a significant increase in individual or cumulative occupational radiation exposure.

PEF has reviewed proposed License Amendment Request #289, Revision 0, and concludes it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with this request.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT C

LICENSE AMENDMENT REQUEST #289, REVISION 0

Proposed Revised Improved Technical Specifications Pages

Strikeout/Shadowed Format

~~Strikeout Text~~ Indicates Deleted Text
Shadowed Text Indicates Added Text

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS-Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72* hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

*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.

3.6 CONTAINMENT SYSTEMS

3.6.6 Reactor Building Spray and Containment Cooling Systems

LCO 3.6.6 Two reactor building spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One reactor building spray train inoperable.	A.1 Restore reactor building spray train to OPERABLE status.	72*hours AND 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours
C. One required containment cooling train inoperable.	C.1 Restore required containment cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

*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.

3.7 PLANT SYSTEMS

3.7.8 Decay Heat Closed Cycle Cooling Water (DC) System

LCO 3.7.8 Two DC trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC train inoperable.	A.1 ------NOTE----------- Enter applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops-MODE 4," for required decay heat removal loops made inoperable by DC train inoperability. --------------------- Restore DC train to OPERABLE status.	72* hours
B. Required Action and associated Completion Time not met.	B.1 Be in Mode 3 AND B.2 Be in Mode 5.	6 hours 36 hours

*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.

3.7 PLANT SYSTEMS

3.7.10 Decay Heat Seawater System

LCO 3.7.10 Two Decay Heat Seawater System trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Decay Heat Seawater System train inoperable.	A.1 --------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops-MODE 4," for required decay heat removal loops made inoperable by Decay Heat Seawater System train inoperability. --------------------- Restore Decay Heat Seawater System train to OPERABLE status.	72* hours
B. Required Action and associated Completion Time not met.	B.1 Be in Mode 3	6 hours
	AND	
	B.2 Be in Mode 5.	36 hours

*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.

BASES

ACTIONS

A.1

With one or more ECCS trains inoperable and at least 100% of the flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72* hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 3) that are based on a risk evaluation and is a reasonable time for many repairs.

*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.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that the safety injection (SI) flow equivalent to 100% of a single train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

For example, removing a train of the recirculation line to the RB sump or the entire bank of valves for maintenance does not render the HPI System inoperable, given the diverse ability to recirculate to the Makeup Tank. HPI satisfies Criterion 3 of the NRC Policy Statement which addresses SSCs that are part of the primary success path, and which function or actuate to mitigate a design basis accident or transient challenging a fission product barrier. Since this recirculation line supports piggyback operation in long-term cooling, and piggyback operation is not a primary success path, LCO 3.5.2 need not be entered when this recirculation path is not available.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 3) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

With one or more components inoperable such that the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

BASES

LCO (continued)

iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two RB spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

Each RB Spray System train includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safeguards Actuation System signal and manually transferring suction to the reactor building emergency sump.

Each Containment Cooling System train includes demisters, cooling coils, dampers, an axial flow fan driven by a two speed water cooled electrical motor, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the RB spray trains and containment cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the RB Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one RB spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72* hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat

(continued)

BASES

ACTIONS

A.1 (continued)

removal capability afforded by the OPERABLE RB spray train and cooling system train(s), reasonable time for repairs, and the low probability of a DBA occurring during this period.

*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 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times", for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable RB spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time to attempt restoration of the RB spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the required containment cooling trains inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RB Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS

A.1 (continued)

With one DC train inoperable, action must be taken to restore the train to OPERABLE status within 72* hours. In this Condition, the remaining OPERABLE DC train is adequate to perform the heat removal function. The 72 hour Completion Time for restoring full DC System OPERABILITY is the same as that for the ECCS Systems, whose safety functions are supported by the DC System. This Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

*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.

B.1 and B.2

If the inoperable DC train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual and power operated valves in the DC flow path provides assurance that the proper flow paths exist for DC operation. The isolation of the DC flow to individual components may render those components inoperable, but does not affect the operability of the DC system. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path.

(continued)

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops - MODE 4," should be entered if an inoperable decay heat seawater train results in an inoperable required DHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for an inoperability of a required DHR loop.

If one of the decay heat seawater trains is inoperable, action must be taken to restore the train to OPERABLE status within 72* hours. In this Condition, the remaining OPERABLE train is adequate to perform the heat removal function. The 72 hour Completion Time for restoring full Decay Heat Seawater System OPERABILITY is the same as that for the ECCS Systems, whose safety functions are supported by the Decay Heat Seawater System. This Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

*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.

B.1 and B.2

If the inoperable decay heat seawater train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Verifying the correct alignment for manual valves in the Decay Heat Seawater System flow path provides assurance that the proper flow paths exist for DC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path.

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PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT D

LICENSE AMENDMENT REQUEST #289, REVISION 0

Proposed Revised Improved Technical Specifications Pages

Revision Bar Format

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS-Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72* hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

*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.

3.6 CONTAINMENT SYSTEMS

3.6.6 Reactor Building Spray and Containment Cooling Systems

LCO 3.6.6 Two reactor building spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One reactor building spray train inoperable.	A.1	Restore reactor building spray train to OPERABLE status.	72* hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	84 hours
C. One required containment cooling train inoperable.	C.1	Restore required containment cooling train to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

*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.

3.8 PLANT SYSTEMS

3.7.8 Decay Heat Closed Cycle Cooling Water (DC) System

LCO 3.7.8 Two DC trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC train inoperable.	A.1 ------NOTE----------- Enter applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops-MODE 4," for required decay heat removal loops made inoperable by DC train inoperability. --------------------- Restore DC train to OPERABLE status.	72* hours
B. Required Action and associated Completion Time not met.	B.1 Be in Mode 3 AND B.2 Be in Mode 5.	6 hours 36 hours

*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.

3.7 PLANT SYSTEMS

3.7.10 Decay Heat Seawater System

LCO 3.7.10 Two Decay Heat Seawater System trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One Decay Heat Seawater System train inoperable.	A.1	--------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops-MODE 4," for required decay heat removal loops made inoperable by Decay Heat Seawater System train inoperability. --------------------- Restore Decay Heat Seawater System train to OPERABLE status.	72* hours
B. Required Action and associated Completion Time not met.	B.1	Be in Mode 3	6 hours
	AND		
	B.2	Be in Mode 5.	36 hours

*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.

BASES

ACTIONS

A.1

With one or more ECCS trains inoperable and at least 100% of the flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72* hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 3) that are based on a risk evaluation and is a reasonable time for many repairs.

*On a one-time basis. an Emeraencv Core Coolina Svstem train mav be inooerable as soecified bv Reauired Action A.1 for uo to 10 davs to allow oerformance of Decav Heat Seawater Svstem Pumo RWP-3B reoairs online. Uoon comoletion of the refurbishment and system restoration this footnote is no longer applicable.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that the safety injection (SI) flow equivalent to 100% of a single train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

For example, removing a train of the recirculation line to the RB sump or the entire bank of valves for maintenance does not render the HPI System inoperable, given the diverse ability to recirculate to the Makeup Tank. HPI satisfies Criterion 3 of the NRC Policy Statement which addresses SSCs that are part of the primary success path, and which function or actuate to mitigate a design basis accident or transient challenging a fission product barrier. Since this recirculation line supports piggyback operation in long-term cooling, and piggyback operation is not a primary success path, LCO 3.5.2 need not be entered when this recirculation path is not available.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 3) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

With one or more components inoperable such that the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

BASES

LCO (continued)	iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two RB spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs. Each RB Spray System train includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safeguards Actuation System signal and manually transferring suction to the reactor building emergency sump. Each Containment Cooling System train includes demisters, cooling coils, dampers, an axial flow fan driven by a two speed water cooled electrical motor, instruments, and controls to ensure an OPERABLE flow path.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the RB spray trains and containment cooling trains. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the RB Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.
ACTIONS	<u>A.1</u> With one RB spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72* hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat

(continued)

BASES

ACTIONS

A.1 (continued)

removal capability afforded by the OPERABLE RB spray train and cooling system train(s), reasonable time for repairs, and the low probability of a DBA occurring during this period.

*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 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times", for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable RB spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time to attempt restoration of the RB spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the required containment cooling trains inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RB Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS

A.1 (continued)

With one DC train inoperable, action must be taken to restore the train to OPERABLE status within 72* hours. In this Condition, the remaining OPERABLE DC train is adequate to perform the heat removal function. The 72 hour Completion Time for restoring full DC System OPERABILITY is the same as that for the ECCS Systems, whose safety functions are supported by the DC System. This Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

*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.

B.1 and B.2

If the inoperable DC train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual and power operated valves in the DC flow path provides assurance that the proper flow paths exist for DC operation. The isolation of the DC flow to individual components may render those components inoperable, but does not affect the operability of the DC system. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path.

(continued)

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops - MODE 4," should be entered if an inoperable decay heat seawater train results in an inoperable required DHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for an inoperability of a required DHR loop.

If one of the decay heat seawater trains is inoperable, action must be taken to restore the train to OPERABLE status within 72* hours. In this Condition, the remaining OPERABLE train is adequate to perform the heat removal function. The 72 hour Completion Time for restoring full Decay Heat Seawater System OPERABILITY is the same as that for the ECCS Systems, whose safety functions are supported by the Decay Heat Seawater System. This Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

*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.

B.1 and B.2

If the inoperable decay heat seawater train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5. within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Verifying the correct alignment for manual valves in the Decay Heat Seawater System flow path provides assurance that the proper flow paths exist for DC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path.

(continued)

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT E

LICENSE AMENDMENT REQUEST #289, REVISION 0

PSA Risk Assessment of RWP-3B Extended AOT

SYSTEM #	N/A
CALC. SUB-TYPE	
PRIORITY CODE	
QUALITY CLASS	Nonsafety

NUCLEAR GENERATION GROUP

P-05-0001

PSA Risk Assessment of RWP-3B Extended AOT

(Title including structures, systems, components)

☐ BNP UNIT

☒ CR3 ☐ HNP ☐ RNP ☐ NES ☐ ALL

APPROVAL

REV	PREPARED BY	REVIEWED BY	SUPERVISOR
0	Signature D. Miskiewicz Name David Miskiewicz Date 1/12/05	Signature Scott Brinkman Name Scott Brinkman Date 1/12/2005	Signature John R. Caves Name John Caves Date 1/12/05

(For Vendor Calculations)

Vendor________________ Vendor Document No. __________

Owner's Review By ________________ Date __________

LIST OF EFFECTIVE PAGES

PAGE	REV	PAGE	REV	ATTACHMENTS		
i-v	0			Number	**Rev**	Number of Pages
1-16	0					
				AMENDMENTS		
				Letter	**Rev**	Number of Pages

Rev. #	Revision Summary (list of ECs incorporated)
0	Initial issue of calculation

Document Indexing Table

Document Type (e.g. CALC, DWG, TAG, PROCEDURE, SOFTWARE)	**ID Number** (e.g., Calc No., Dwg. No., Equip. Tag No., Procedure No., Software name and version)	**Function** (i.e. IN for design inputs or references; OUT for affected documents)	**Relationship to Calc.** (e.g. design input, assumption basis, reference, document affected by results)
CALC	P-02-0001, Rev 1	IN	Design input, Reference

Record of Lead Review

Design P05-0001, **Revision** 0

The signature below of the Lead Reviewer records that:
- the review indicated below has been performed by the Lead Reviewer;
- appropriate reviews were performed and errors/deficiencies (for all reviews performed) have been resolved and these records are included in the design package;
- the review was performed in accordance with EGR-NGGC-0003.

☐ **Design Verification Review** ☒ **Engineering Review** ☐ **Owner's Review**

☐ Design Review
☐ Alternate Calculation
☐ Qualification Testing

☐ **Special Engineering Review** ______

☐ YES ☐ N/A **Other Records are attached**

Scott A Brinkman	[signature]	PSA	1-12-2005
Lead Reviewer (print)	**(sign)**	**Discipline**	**Date**

Item No.	Deficiency	Resolution
1)	Table 4, zone AB-119-6A, appears to be missing source HY-6A, 8.00E-4	Added HY-6A.
2)		
3)		
4)		
5)		
6)		

FORM EGR-NGGC-0003-2-5
This form is a QA Record when completed and included with a completed design package. Owner's Reviews may be processed as stand alone QA records when Owner's Review is completed

Record of Interdisciplinary Reviews

PART I — DESIGN ASSUMPTION / INPUT REVIEW: APPLICABLE ☒ Yes ☐ No

The following organizations have reviewed and concur with the design assumptions and inputs used in this calculation:

	Name	Signature	Date
Engineering	KRCAMPBELL	[signature]	1/11/05
~~Operations~~ Other dam 1/11/05			
~~Licensing~~ dam 1/11/05			

PART II — RESULTS REVIEW:

The following organizations are aware of the impact of the results of this calculation (on designs, programs and procedures):

	Name	Signature	Date
Engineering ☒ Yes ☐ NO	KRCAMPBELL	[signature]	1/11/05

Comments:

	Name	Signature	Date
Operations ☑ Yes ☐ NO	B. Wunderly	[signature]	1/11/05

Comments:

	Name	Signature	Date
Licensing	Loretta V Cecilia	[signature]	1/11/05

Comments:

	Name	Signature	Date
Other APPENDIX E Review	PABLO M. RUBIO	[signature]	1/12/05

Comments:

TABLE OF CONTENTS

1.0 Purpose

This calculation assesses the risk associated with increasing the Improved Technical Specification (ITS) allowed outage time (one time from 72 hours to 10 days) in order to perform maintenance to RWP-3B. This assessment will include the increase in core damage frequency (CDF) and large early release frequency (LERF) associated with the activity and recommended compensatory actions which can be used to minimize the risk.

2.0 References

1. CR3 calculation P-02-0001, Rev.1, "CR3 PSA Model of Record - MOR_03a"
2. NRC RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis"
3. NRC RG 1.177, "An Approach for Plant-Specific Risk-Informed Decision Making: Technical Specifications"
4. CR3 Individual Plant Examination of External Events (IPEEE) Revision 1, March 1997
5. CR3 Response to IPEEE NRC RAI, dated March 28, 2000
6. CR3 Fire Study

3.0 Design Inputs

This calculation is not a design basis analysis. The intent is to represent the as designed and operated plant in a realistic manner for the purpose of assessing the risk of core damage for different scenarios. The primary input for performing this assessment is the CR3 Probabilistic Risk Assessment (PRA) Model of Record and supporting documentation (Ref. 1) and the scenario of interest (i.e. – 10 day AOT for RWP-3B).

4.0 Assumptions

The primary assumption for this assessment is that the plant will be operated in accordance with existing procedures and practices consistent with reference 1.

5.0 Calculation / Analysis Details

5.1 Internal Events (including flooding)

The internal events risk assessment for this task was performed using EOOS with the current CR3 model of record (Ref. 1). Several cases were run and the results are provided in Table 1. CR3 has a diverse support system arrangement for the makeup/HPI system. Because the RW pump being analyzed (RWP-3B) provides normal support to makeup pump (MUP-1C), alternate possible plant configurations are considered.

The "normal" or historically preferred configuration for CR3 is:

- MUP-1B is running powered from "A", and ES selected
- MUP-1C is ES standby and cooled from DHCCC (and RWP-3B)
- MUP-1A is not ES selected, but available and cooled from NSCCC
- ES 4160 "A" is powered from the OPT
- ES 4160 "B" is powered from the BEST
- RWP-1 and SWP-1C are the normally running cooling pumps

When RWP-3B is out of service cooling to MUP-1C is only available from NSCCC. By selecting MUP-1A instead of MUP-1C as the ES standby pump some additional diversity of MUP cooling can be maintained since MUP-1A can still be cooled from either NSCCC or DHCCC.

Sensitivity cases were also developed based on increasing the probability of loss of offsite power and known model changes planned for the next update. These changes are listed in Table 2.

The results from all cases considered are compared to the risk thresholds presented in Regulatory Guide 1.174 and 1.177 (Ref. 2 and 3).

The impacts due to external events such as fires, and severe weather are primarily evaluated qualitatively to provide risk insights.

Based on the analysis, compensatory actions which can help minimize the increase in risk associated with the extended AOT are also provided.

Table 1 –Cases Run for RWP-3B Maintenance Activity

Case	Configuration	CDF	CDFI	dCDFI (/yr)	ICCDP (7d)	ICCDP (10d)	Days to 1E-06	LERF	LERFI	dLERFi (/yr)	ICLERP (10d)
1	Current Model of Record (1)	7.49E-06						3.72E-07			
2	RWP-3B OOS, MUP-1C (SW)		6.32E-05	5.57E-05	1.1E-06	1.5E-06	6		4.66E-07	9.4E-08	2.6E-09
3	RWP-3B, MUP-1A/B ES		5.79E-05	5.04E-05	9.7E-07	1.4E-06	7				
4	RWP-3B, MUP-1A/B ES, A (DC) (5)		4.84E-05	4.09E-05	7.8E-07	1.1E-06	8				
5	Case 3, no maintenance (2)		5.50E-05	4.75E-05	9.1E-07	1.3E-06	7				
6	Case 4, no maintenance (2,5)		4.58E-05	3.83E-05	7.3E-07	1.0E-06	9				
7	Case 5, loop x3 (3)		5.74E-05	4.99E-05	9.6E-07	1.4E-06	7				
8	Revised MOR (4)	5.88E-06									
9	Case 5, Revised MOR (4)		3.96E-05	3.37E-05	6.5E-07	9.2E-07	10				
10	Case 6, Revised MOR (4)		3.03E-05	2.44E-05	4.7E-07	6.7E-07	14				

Notes	
(1)	Internal Events w/average maintenance @ 1E-10, alignments : MUP-1B running, MUP-1B/1C ES selected, MUP-1C cooled by DHCCC, MUP-1A cooled by NSCCC. HVAC "A" running, RWP-1running, SWP-1C running, ES 4160 "A" = OPT, ES 4160 "B" = BEST
(2)	Maintenance events for FWP-7, EFP, DHP, DCP, RWP, SWP, EDG, MUP set to zero
(3)	Both LOOP initiators (T3 and T15) increased by a factor of 3
(4)	Includes model changes currently planned for next update (see Table 2). The most significant change is the deletion of DHV-110/111 miscalibration as a failure mode for high pressure recirculation
(5)	Aligning MUP-1A to DC cooling introduces a concern based on Appendix R considerations, and should only be used if necessary. (see OP-402, Note preceding step 4.2.10)

The data in Table 1 can be interpreted as follows:

Case –	The identification number for the specific configuration analyzed
Configuration –	Description of the case analyzed
CDF/LERF–	These are baseline values based on quantification at 1E-10 truncation, and including average maintenance probabilities.
CDFi/LERFi–	This are the instantaneous values as determined by adjusting the model for the specific case and quantifying
dCDFi –	This is the delta CDF for the case (= $CDFi_{case} - CDF_{baseline}$)
dLERFi –	This is the delta LERF for the case (= $LERFi_{case} - LERF_{baseline}$)
ICCDP (time) –	This is the Incremental Conditional Core Damage Probability for the case and a specified time period (= $dCDFi_{case}$ * (time/1yr)). For this evaluation, this value can also be interpreted as the delta CDF (dCDF) for the time specified
ICLERP (time) –	This is the Incremental Conditional Large Early Release Probability for the case and a specified time period (= $dLERFi_{case}$ * (time/1yr)) For this evaluation, this value can also be interpreted as the delta LERF (dLERF) for the time specified
Days to 1E-06 –	This the number of days the configuration can remain before reaching a delta CDF of 1E-06 for the case configuration. This is the limit specified by RG 1.174 as a very small change. (= (1E-06 * dCDFi) / 365 days)

Table 2 (model changes to MOR_03a)

File	Description of Change	Comment
B011	Deleted PAVM412C	Redundant to event in P723
H2825	Add H005	Block MUP start if cooling OOS
H2725	Add S305	Cont.
H2625	Add H001	Cont.
H011	Delete IE_T10FT, Add H014	Cont.
H014	New "OR" IE_10FT, IE_T11	Cont.
H013	Delete IE_T11	Cont.
RHURCPTY	Set to 5.4E-02	HRA update
Z_HBESS Z_HCESS Z_HAESS	Added operator ES override for EOOS alignment	Formula did not allow operators to override ES preferences
Z_HBESS Z_HCESS Z_HAESS	Added operator ES override for EOOS alignment	Formula did not allow operators to override ES preferences
Z_HBESS Z_HCESS Z_HAESS	Added operator ES override for EOOS alignment	Formula did not allow operators to override ES preferences
LMMTRNAX	Not a module – renamed as gate L070	DHV-110 misalignments not applicable in HPR cases
L355	Add new gate L071 "OR" LHUDH21X, LHUD210X delete L070	Remove DHV-110 events from failing HPR
LMMTRNBX	Not a module – renamed as gate L080	DHV-111 misalignments not applicable in HPR cases
L356	Add new gate L081 "OR" LHUDH32X, LHUD211X delete L080	Remove DHV-111 events from failing HPR
HINJA1ML	Rename to SPLT_MA1	Naming convention for splits
HINJA2ML	Rename to SPLT_MA2	Naming convention for splits
HINJB1ML	Rename to SPLT_MB1	Naming convention for splits
HINJB2ML	Rename to SPLT_MB2	Naming convention for splits
HINJA1SL	Rename to SPLT_SA1	Naming convention for splits
HINJA2SL	Rename to SPLT_SA2	Naming convention for splits
HINJB1SL	Rename to SPLT_SB1	Naming convention for splits
HINJB2SL	Rename to SPLT_SB2	Naming convention for splits

5.2 External Events

The CR3 IPEEE and supporting data was reviewed to identify external event influences to the risk for the subject activities. The only potentially significant external events are fires and severe weather. In general weather is difficult to predict, however, the chances of severe weather are greater in Florida during the summer months, and the main impact is an increased probability for loss of offsite power. A sensitivity case (case 7) was performed and demonstrated a minimal increase in risk due to a higher loss of offsite power frequency during the extended AOT.

Table 3 lists the fire zones identified as containing circuits applicable to the RW-DC pumps and there supported front line systems. Because RW is a support system, it is useful to consider the front line systems in this evaluation also. Therefore, the Decay Heat Removal (DH) pumps and the DH Closed Cycle Cooling (DC) pumps were also included in the fire assessment. Table 3 displays the fire areas identified by the CR3 Appendix R fire study which are important. If the fire can be expected to impact both trains or only the "B" train, or manual actions are credited in the Fire Study, then the delta risk is expected to be minimal. The greatest risk impact due to RWP-3B being out of service is expected for fires which impact only the "A" train equipment. These zones are indicated with a "yes".

Table 4 provides the ignition sources and raw frequencies (per the IPEEE) for each fire zone indicated as a candidate for PSA fire risk impact ("yes"). The compensated frequency column eliminates the contribution from transient sources and equipment which will not be operated without special precautions. These sources are shaded or italicized. If automatic suppression exists a credit of 0.1 was applied. Finally a CCDP of 0.1 was applied as a sensitivity for the purpose of estimating a core damage frequency. The delta CDF due to fire based on this table, with RWP-3B out of service, is 9.92E-05-04/yr. This translates in to a delta risk associated with RWP-3B AOT and fire of:

9.92E-05/yr * 10days * 1yr/365days = 2.72E-06

Additional compensatory actions such as dedicated fire watches could be used to further reduce this value.

Table 5 summarizes potential compensatory actions based on this assessment which can be used to reduce the risk of core damage. Other factors (such as personnel safety or practicality) may impact the use of some of these actions.

Table 3 – RW-DC Pump Related Fire Zones

ZONE	RWP-3A	DCP-1A	DHP-1A	RWP-3B	DCP-1B	DHP-1B	Fire Risk Impact due to RWP AOT
AB-75-4						x	neg
AB-75-5			x				yes
AB-95-3AA	x	xR	x				yes
AB-95-3B	xW	xR	xW	xW	xR	xW	minimal
AB-95-3C			x				yes
AB-95-3D				xW		xW	neg
AB-95-3E	x		x				yes
AB-95-3F	x		x				yes
AB-95-3G	x	xR		xW	xR	xW	yes (1)
AB-95-3K	x		x	xW		xW	yes
AB-95-3L		xR					yes
AB-95-3M		xR					yes
AB-95-3N		xR					yes
AB-95-3P		xR					yes
AB-95-3Q		xR					yes
AB-95-3R		xR					yes
AB-95-3T	x	xR					yes
AB-95-3U	x	xR					yes
AB-95-3W	x	xR	x				yes
AB-95-3X				xW	xR	xW	neg
AB-95-3Y			x			xR	yes (1)
AB-95-3Z	x	xR		xW	xR		yes (1)
AB-119-6A		x			xW		yes
AB-119-6E		x					yes
CC-95-101				x	x	x	neg
CC-108-102	x	x	x	xW	xW	xW	yes
CC-108-103			xW	x	x	x	neg
CC-108-104	x	x	x				yes
CC-108-105	xW	xW	xW	x	x	x	neg
CC-108-106	x	x	x	xM		xM	yes
CC-108-107				x	x	x	neg
CC-108-108	x	x	x				yes
CC-108-109	xM	xM	xM	xM	x	xM	minimal
CC-108-110	x	x	x				yes
CC-124-111	x		x	xM	xW	xM	yes
CC-124-115				x	x	x	neg
CC-124-116				x	x	x	neg
CC-124-117	x	x	x				yes
CC-134-118A	xT	xT	xT	xT	xT	xT	minimal
CC-145-118B	xT	xT	xT	xT	xT	xT	minimal
CC-164-121	xT	xT	xT	xT	xT	xT	minimal

x -indicates equipment not available due to fire
xW -indicates protected with fire wrap
xM -indicates available with manual actions
xT -indicates available from remote shutdown panel
xR -Appendix R credits equipment repair for long term availability
(1) -yes based on Appendix R (hardware repair). PSA does not typically credit repair and will classify as minimal for additional analysis

Table 4 – Ignition Sources/Frequencies for Impacted Areas

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
AB-75-5	TRANS-5	9.73E-05					
	BSP-1A	5.21E-05					
	DHP-1A	5.21E-05	2.02E-04	0.00E+00	1	0.1	0.00E+00
AB-95-3AA	TRANS-3AA	9.73E-05					
	MUP-1B	5.21E-05	1.49E-04	5.21E-05	0.1	0.1	5.21E-07
AB-95-3C	MCC MUV-23/24	4.48E-06					
	MCC MUV-25/26	4.48E-06					
	TRANS-3C	9.73E-05	1.06E-04	8.96E-06	0.1	0.1	8.96E-08
AB-95-3E	*MUP-1A*	5.21E-05					
	TRANS-3E	9.73E-05	1.49E-04	0.00E+00	0.1	0.1	0.00E+00
AB-95-3F	*MUP-1C*	5.21E-05			1	0.1	0.00E+00
	TRANS-3F	9.73E-05	1.49E-04	0.00E+00	1	0.1	0.00E+00
AB-95-3K	TRANS-3K	9.73E-05	9.73E-05	0.00E+00	1	0.1	0.00E+00
AB-95-3L	TRANS-3L	9.73E-05					
	ASP-2B	5.21E-05					
	ASP-2A	5.21E-05	2.02E-04	0.00E+00	1	0.1	0.00E+00
AB-95-3M	TRANS-3M	9.73E-05	9.73E-05	0.00E+00	1	0.1	0.00E+00
AB-95-3N	TRANS-3N	9.73E-05	9.73E-05	0.00E+00	1	0.1	0.00E+00
AB-95-3P	*WDP-12A*	5.21E-05					
	WDP-12B	5.21E-05					
	WDP-13A	5.21E-05					
	WDP-13B	5.21E-05					
	TRANS-3P	9.73E-05	3.06E-04	0.00E+00	1	0.1	0.00E+00
AB-95-3Q	TRANS-3Q	9.73E-05	9.73E-05	0.00E+00	1	0.1	0.00E+00
AB-95-3R	TRANS-3R	9.73E-05					
	HTTR-4A	1.20E-05					
	HAYES CAB	4.48E-06					
	HTDP-1B	4.48E-06					
	HTDP-4A	4.48E-06					
	HTTR-1A	1.20E-05					
	HTTR-1B	1.20E-05					
	HTDP-1A	4.48E-06					
	WDP-1B	6.71E-05	2.18E-04	1.21E-04	1	0.1	1.21E-05
AB-95-3T	TRANS-3T	9.73E-05	9.73E-05	0.00E+00	1	0.1	0.00E+00
AB-95-3U	TRANS-3U	9.73E-05	9.73E-05	0.00E+00	1	0.1	0.00E+00
AB-95-3W	*WDP-5C*	5.21E-05					
	WDP-5B	5.21E-05					
	WDP-5A	5.21E-05					
	TRANS-3W	9.73E-05	2.54E-04	0.00E+00	1	0.1	0.00E+00
AB-119-6A	HY-6A	8.00E-04					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	TRANS-6A-H	1.22E-05					
	TRANS-6A-G	1.22E-05					
	TRANS-6A-F	1.22E-05					
	TRANS-6A-E	1.22E-05					
	TRANS-6A-D	1.22E-05					
	TRANS-6A-B	1.22E-05					
	TRANS-6A-A	1.22E-05					
	TRANS-6A-C	1.22E-05	8.98E-04	8.00E-04	0.1	0.1	8.00E-06
AB-119-6E	MTMC-4 R9	4.48E-06					
	MTMC-4 R10	4.48E-06					
	MTMC-4 R11	4.48E-06					
	MTMC-4 R2	4.48E-06					
	MTMC-4 R4	4.48E-06					
	MTMC-4 R6	4.48E-06					
	MTMC-4 R8	4.48E-06					
	MTMC-4 R1	4.48E-06					
	TRANS-6E	9.73E-05					
	MTMC-4 R3	4.48E-06					
	MTMC-4 R7	4.48E-06					
	HY-6E	8.00E-04					
	MTMC-4 R5	4.48E-06					
	MTMC-21 R1	4.48E-06					
	MTMC-21 R2	4.48E-06					
	MTMC-21 R3	4.48E-06					
	MTMC-21 R6	4.48E-06					
	MTMC-21 R4	4.48E-06					
	MTMC-21 R5	4.48E-06	9.73E-04	8.76E-04	0.1	0.1	8.76E-06
CC-108-102	TRANS-102	9.73E-05					
	AHF-69	1.85E-05					
	REMOTE SHUTDOWN PNL	4.48E-06	1.20E-04	2.30E-05	1	0.1	2.30E-06
CC-108-104	TRANS-104	9.73E-05	9.73E-05	0.00E+00	1	0.1	0.00E+00
CC-108-106	TRANS-106-A	9.73E-05					
	DPBC-1A	8.89E-05					
	DPBC-1E	8.89E-05					
	DPDP-1A	4.48E-06					
	DPBC-1C	8.89E-05	3.68E-04	2.71E-04	1	0.1	2.71E-05
CC-108-108	MTSW-2C R4	7.20E-06					
	AHF-72	1.85E-05					
	MTTR-4	7.20E-06					
	MTSW-2D R6	7.20E-06					
	MTSW-2D R7	7.20E-06					
	MTSW-2D R5	7.20E-06					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	CAIT-1	7.20E-06					
	MTSW-2D R3	7.20E-06					
	MTSW-2D R2	7.20E-06					
	MTSW-2D R1	7.20E-06					
	MTSW-2C R6	7.20E-06					
	MTSW-2C R3	7.20E-06					
	MTSW-2C R2	7.20E-06					
	MTSW-2C R1	7.20E-06					
	DPDP-8A	7.20E-06					
	MTSW-2C R7	7.20E-06					
	RSD RLY A	7.20E-06					
	MTSW-2D R4	7.20E-06					
	RSD AUX A RLY	7.20E-06					
	RCMP-3A	7.20E-06					
	RSD RLY A1	7.20E-06					
	TRANS-108-B	4.86E-05					
	TRANS-108-A	4.86E-05	2.60E-04	1.63E-04	1	0.1	1.63E-05
CC-108-110	VBIT-1A	4.48E-06					
	VBIT-1C	4.48E-06					
	VBXS-3A	4.48E-06					
	VBXS-1A	4.48E-06					
	VBXS-1C	4.48E-06					
	VBTR-3C	1.20E-05					
	VBTR-3A	1.20E-05					
	VBXS-3C	4.48E-06					
	VBDP-12	4.48E-06					
	TRANS-110	9.73E-05					
	VBTR-2C	1.20E-05					
	AHHE-55	4.48E-06					
	AHHE-54	4.48E-06					
	VBDP-13	4.48E-06					
	VBTR-2A	1.20E-05	1.90E-04	9.28E-05	1	0.1	9.28E-06
CC-124-111	DRRD7-6A	4.48E-06					
	DRRD3-8	4.48E-06					
	DRRD7-5A	4.48E-06					
	DRRD6B	4.48E-06					
	DRRD7-7A	4.48E-06					
	DRRD7-6B	4.48E-06					
	DRRD6A	4.48E-06					
	DRRD7-5B	4.48E-06					
	DRRD4-1	4.48E-06					
	DRRD2-3	4.48E-06					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	DRRD3-1	4.48E-06					
	DRRD3-2	4.48E-06					
	DRRD3-3	4.48E-06					
	DRRD3-4	4.48E-06					
	DRRD3-5	4.48E-06					
	DRRD3-6	4.48E-06					
	DRRD4-2	4.48E-06					
	DRTR-1B	1.20E-05					
	DRRD5R	4.48E-06					
	DRRD7-7B	4.48E-06					
	DRRD4-3	4.48E-06					
	DRRD4-4	4.48E-06					
	DRRD4-5	4.48E-06					
	DRRD4-7	4.48E-06					
	LIGHTING XFMR A	1.20E-05					
	DRRD5L	4.48E-06					
	DRRD3-7	4.48E-06					
	TRANS-111-L	6.49E-06					
	DRRD8B	4.48E-06					
	TRANS-111-F	6.49E-06					
	TRANS-111-G	6.49E-06					
	TRANS-111-H	6.49E-06					
	TRANS-111-I	6.49E-06					
	TRANS-111-D	6.49E-06					
	TRANS-111-K	6.49E-06					
	TRANS-111-C	6.49E-06					
	TRANS-111-M	6.49E-06					
	TRANS-111-N	6.49E-06					
	TRANS-111-O	6.49E-06					
	LIGHTING XFMR B	1.20E-05					
	MUX 4	4.48E-06					
	TRANSMITTER PWR SUPP CAB A,AB, B	4.48E-06					
	TRANS-111-J	6.49E-06					
	DRTR-1A	1.20E-05					
	DRRD7-8B	4.48E-06					
	DRRD8A	4.48E-06					
	VBTR-1A	1.20E-05					
	VBTR-1B	1.20E-05					
	RRHV	4.48E-06					
	TRANS-111-E	6.49E-06					
	DRRD2-2	4.48E-06					
	DRRD7-8A	4.48E-06					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	DRRD4-6	4.48E-06					
	EHCC-1	4.48E-06					
	EHCC-2	4.48E-06					
	EHCC-3	4.48E-06					
	TRANS-111-A	6.49E-06					
	TRANS-111-B	6.49E-06					
	RRPSA	4.48E-06					
	PAX CAB	4.48E-06					
	MUX 2	4.48E-06					
	CDR VOLTAGE REG B	4.48E-06					
	CRDM GROUP POWER SUPPLY CAB	4.48E-06					
	MUX 1	4.48E-06					
	MUX 5	4.48E-06					
	DRRD4-8	4.48E-06					
	ACTR (NEAR JAIL DOOR)	1.20E-05					
	DRRD2-1	4.48E-06					
	CDR VOLTAGE REG A	4.48E-06					
	COMM CAB (PAX)	4.48E-06					
	DPDP-4B	4.48E-06					
	DPDP-4A	4.48E-06					
	COMTEL 2020 REMOTE	4.48E-06					
	CRD BKR A CAB	4.48E-06					
	CRD BKR B CAB	4.48E-06					
	ACTR (SW CORNER)	1.20E-05					
	RR3A	4.48E-06					
	AHF-54A	1.85E-05					
	AHDP-11	4.48E-06					
	RR3B	4.48E-06					
	RR3	4.48E-06					
	ACTR-15	1.20E-05					
	RR2AB	4.48E-06					
	RR1B	4.48E-06					
	RR1	4.48E-06					
	RFM	4.48E-06					
	MUX 3	4.48E-06					
	RR2	4.48E-06	5.06E-04	4.09E-04	0.1	0.1	4.09E-06
CC-124-117	TRANS-117-B	1.39E-05					
	MTSW-3F R1	7.20E-06					
	MTSW-3F R3	7.20E-06					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	RC RCITS-C	7.20E-06					
	TRANS-117-G	1.39E-05					
	MTSW-3F R2	7.20E-06					
	DPDP-5A	7.20E-06					
	MTSW-3F	1.20E-05					
	TRANS-117-A	1.39E-05					
	AHF-75	1.85E-05					
	AHF-74	1.85E-05					
	ES MCC 3AB/TS	7.20E-06					
	TRANS-117-C	1.39E-05					
	TRANS-117-D	1.39E-05					
	RC RCITS-A	7.20E-06					
	DPDP-8C	7.20E-06					
	TRANS-117-E	1.39E-05					
	TRANS-117-F	1.39E-05	2.04E-04	1.07E-04	1	0.1	1.07E-05

5.3 Sensitivity Cases

Several sensitivity cases were evaluated. First the frequency of losing offsite power was increased to assess the impact of severe weather (Case 7 in Table 1). This involved increasing both the normal LOOP and partial LOOP initiating events (T3, T15) in the PRA model. The increase was not significant.

Because the PRA models are "living" they are constantly being updated and improved. Several changes are planned for the next update and were applied to assess potential impacts. The specific changes are listed in Table 2. In each of the cases (cases 8-10 in Table 1), the delta risk for the extended AOT is lower with changes are implemented. This indicates that the current assessment is conservative.

Current licensing conditions at CR3 require transition to mode 5 if RWP-3B is inoperable for more than 72 hours. Because the RW-DC system is vital for shutdown cooling there is also some increase in risk if this activity were performed while shutdown. Performing this activity with a "hot" RCS provides more options for the use of secondary cooling for RCS heat removal. CR3s current PRA does not quantify the specific risk involved, however, it is clear that the current assessment is conservative in this respect.

5.4 Compensatory Actions

Based on the risk assessment of the extended AOT, the increase in risk warrants that compensatory actions should be implemented which can reduce the risk by lowing the likelihood of initiating events such as LOOP or fire, and by increasing the likelihood of successful mitigation by optimizing the plant configuration, ensuring availability of the operational equipment, and enhancing operator awareness. Table 5 list specific items which should be considered.

Table 5 – Potential Compensatory Actions

Item	Discussion	Credited in CDF
Limit maintenance beyond RWP-3B	Normal (a)(4) assessments will be used. Maintenance activities which will increase risk beyond acceptable limits will be re-scheduled	The assessment assumes zero maintenance on selected important equipment.
Consider alternate makeup pump configurations	Depending plant configuration, the diversity of available support options can be increased.	This action can have significant effect, but should be evaluated in combination with all actions considered.
Walkdowns / validation of the operable ("A") train equipment as practical.	Provides additional qualitative assurance that the available equipment will perform as required. SP-300 can be referenced.	No probabilistic credit is given in the evaluation for these activities.
Pre-job discussions on the impact not having RWP-3B during an event and potential recovery options such as cross-tying MUP suction.	Piping configurations allow the use of DHP-1A to provide a suction source to MUP-1C however this is not proceduralized for this application..	The PRA does not credit this action in very many scenarios, however, if the probability of this action is reduced there is still a small benefit.
Establish fire watches in the zones indicated "yes" in table 3 based on PSA and Appendix R considerations, to limit fire initiators and combustibles. In some cases enhanced manual suppression may be used.	Limit activities associated with initiation of a fire (welding/grinding/etc., operating standby equipment) or storage of transient combustibles.	A sensitivity type of analysis was performed for this calculation which shows the risk of fires in to be a significant contributor. Credit was added to compensate for monitoring transient combustibles and avoiding the use of standby or normally unused equipment.

6.0 Results / Conclusions

The PSA risk associated with the activity to repair RWP-3B is reasonable to support a one time on-line AOT extension request for 10 days based on ICCDP and ICLERP. The evaluation assumes no other equipment beyond the evaluated systems will be removed from service if the risk is adversely impacted based on maintenance rule (a)(4) risk assessments, which will be performed before and during the activity. Additional compensatory actions are provided in Table 5 which can further reduce the risk if practical. Their use should be based on the specific plant configuration during the use of the extended AOT.

The bounding risk increase for this activity due to internal events is estimated with an ICCDP of 1.5E-06 (case 2) and a sensitivity analysis provides an estimated risk due to fires of 2.72E-06. This is well below 1E-05 and is considered a small increase per RG 1.174. Also, the ICLERP of 2.60E-09 is well below the RG 1.174 limit of 1E-07 to be considered very small. These risks can be further reduced by limiting maintenance on other risk significant equipment listed in Note 2 for Table 1, however, the RG 1.174 classification is unchanged

The detailed results show a range of risk values for the various configurations based on a 10 day AOT. The lowest risk is shown by case 6 (ICDP = 1.0E-06) which is based on aligning MUP-1A up the DHCCC for cooling and limiting maintenance to unaffected equipment. While possible there is an Appendix R precaution with this configuration. Leaving MUP-1A on NSCCC (case 5) is only slightly higher (ICDP = 1.3E-06) and avoids the potential concern. Both of these configurations place this activity as a small risk per RG 1.174.

A sensitivity case was run to assess the impact of increasing the loss of offsite power frequency (case 7). Tripling the frequency did not significantly increase the risk. Additional sensitivity cases (8-10) were run to assess the impact of future planned model changes. In each of these cases the risk impact was lower than with the current model. In case 10 the risk was below 1E-06. Additionally, there is some increased risk to performing this activity while shutdown in mode 5, which will further reduce the total delta risk of performing this activity at power.

Based on the IPEEE, fire can be a significant contributor to risk, however as shown, the risk can be estimated to be in the small risk region as defined by RG 1.174. In order to minimize the potential impact, compensatory actions can be used to reduce the probability of a fire occurring and enhance fire detection and suppression in the more vulnerable areas.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT F

LICENSE AMENDMENT REQUEST #289, REVISION 0

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Progress Energy Florida (PEF) in this document. Any other actions discussed in the submittal represent intended or planned actions by PEF. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing and Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

<table>
<tr><th>Commitment</th><th>Due Date</th></tr>
<tr><td>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.</td><td rowspan="8">During one-time extended (greater than 72 hours) RWP-3B Maintenance</td></tr>
<tr><td>CR-3 will select beneficial Makeup Pump configurations.</td></tr>
<tr><td>Operator attention to the importance of protecting the operable redundant train and support systems will be increased.</td></tr>
<tr><td>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 Operating Procedure (EOP-14), Enclosure 7, Emergency Feedwater Pump (EFWP) Management.</td></tr>
<tr><td>CR-3 will not schedule elective maintenance in the switchyard that would challenge the availability of offsite power.</td></tr>
<tr><td>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.</td></tr>
<tr><td>CR-3 will not initiate an extended RWP-3B maintenance outage if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.</td></tr>
<tr><td>CR-3 will evaluate the material condition of the redundant train to ensure that there is no negative trend that could challenge operability.</td></tr>
<tr><td>CR-3 will submit supplemental information regarding the risk significant fire zones including additional specific compensatory measures by February 11, 2005.</td><td>February 11, 2005</td></tr>
</table>