Progress Energy

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

Ref: 10 CFR 50.90

December 9, 2003
3F1203-03

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

Subject: Crystal River Unit 3 – License Amendment Request #281, Revision 0
Revised Improved Technical Specification (ITS) 3.7.5, Emergency Feedwater System

Dear Sir:

Progress Energy Florida, Inc. (PEF) hereby submits License Amendment Request (LAR) #281, Revision 0, which requests a change to the Crystal River Unit 3 (CR-3) Facility Operating License in accordance with 10 CFR 50.90. LAR #281 revises Technical Specifications (ITS) 3.7.5 to allow a one-time increase in the Completion Time for restoring an inoperable Emergency Feedwater System train to Operable status. The proposed change is being submitted to allow the realignment of the diesel-driven Emergency Feedwater Pump (EFP-3) during power operations.

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

PEF respectfully requests that this LAR be noticed in the Federal Register as soon as practical and the review of this LAR performed to support an approval date of October 1, 2004.

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

No new regulatory commitments are made in this letter.

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

A001

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

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/pei

Attachments:

A. Background, Description of Proposed Change, Reason for Request, and Evaluation of Request
B. Regulatory Analysis
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 EFP-3 Extended AOT

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 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 9th day of December, 2003, 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 #281, REVISION 0

Background, Description of Proposed Change, Reason for Request, and Evaluation of Request

Background

The Crystal River Unit 3 (CR-3) Emergency Feedwater (EFW) System contains two Emergency Feedwater Pumps (EFP) required by Improved Technical Specification (ITS) (EFP-3 and EFP-2). EFP-3 is a diesel-driven pump and EFP-2 is a turbine driven pump. EFP-3 has been exhibiting an elevated vibration on the speed increaser located between the diesel engine and the pump. Although the pump is currently operable, prudent action dictates that corrective maintenance (realignment) should be performed to reduce the vibration prior to the next refueling outage scheduled for Fall 2005. The preferred time to perform the maintenance would be fourth quarter 2004.

ITS 3.7.5, "Emergency Feedwater (EFW) System," requires that two EFW System trains shall be OPERABLE. If one train is inoperable, Condition "B" allows operation to continue for 72 hours. The estimated time required to perform the proposed maintenance activity is 7 days. Based on Institute of Nuclear Operations (INPO) recommended practice, maintenance activities are not normally scheduled for more than half of the allowed Completion Time. Therefore, a one-time extension of the ITS 3.7.5 Completion Time to 14 days is requested in order to perform the realignment online

Description of the Proposed License Amendment Request

This one-time License Amendment Request (LAR) #281, Revision 0, is proposing to add a NOTE to the 72 hour and 10 day Completion Times of ITS 3.7.5, Condition B, Required Action B.1, as follows:

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

The ITS Bases for 3.7.5, Action B.1, will be revised as follows:

> "If one of the EFW trains is inoperable, action must be taken to restore the train to OPERABLE status within 72* hours...
>
> The 10 day* Completion Time for Required Action B.1..."

The following NOTE will be added to the ITS Bases for 3.7.5, Action B.1:

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

Reason for Request

As explained above, in-service testing of the EFP-3 speed increaser shows an elevated vibration level. Since the duration of the repair activity is greater than the 72 hour Completion Time specified in ITS 3.7.5, the repair work must be done in a shutdown MODE unless a one-time extension of the Completion Time for up to 14 days is approved. Thus, approval of the proposed

LAR will allow the performance of the repair online, and will prevent a potential forced shutdown.

Evaluation of Request

System Description

The EFW and Auxiliary Feedwater (AFW) systems are shown in simplified schematic in Figure 1. The EFW system has one 100% capacity turbine-driven Emergency Feedwater pump and one 100% capacity diesel-driven Emergency Feedwater pump. These pumps are automatically started and are controlled by the Emergency Feedwater Initiation and Control (EFIC) system. In addition, a safety–grade 100% capacity motor driven pump, EFP-1, is available for manual initiation and can be manually loaded on the "A" train emergency diesel generator if electrical loading capacity is available. As additional defense-in-depth, the AFW system motor-driven pump (FWP-7) (500 gpm capacity at a Once Through Steam Generator (OTSG) pressure of 1100 psig) can be manually initiated and is powered by offsite power or a non-safety related diesel backup power supply (MTDG-1).

The EFW system supplies flow to the OTSGs when main feedwater is not available. The EFW system is relied upon for accident mitigation for events including small-break loss-of-coolant accidents, loss of main feedwater, loss of offsite power (LOOP), main feedwater line break, main steam line break, and Anticipated Transient Without Scram (ATWS). Any of the three EFW pumps can provide more than the required 550 gpm (at an OTSG pressure of 1082 psig) flow to the OTSGs needed to mitigate the most limiting of these events.

FWP-7 is manually initiated and controlled, independently of the EFIC System. There is minimal sharing of flow paths, components, and water sources between AFW and the Emergency Feedwater (EF) System. The AFW pump is manually operated from the main control room. Controls and indicators for the FWP-7 and AFW flow control valves are included on the main control board.

The preferred water source for both EFW trains is the Seismic 1, tornado-hardened, dedicated Emergency Feedwater Tank (EFT-2) with a surveilled capacity of 150,000 useable gallons. Preferred backup supplies of emergency feedwater are provided by the non-safety related, Seismic 1, Condensate Storage Tank (CDT-1) with a surveilled capacity of 120,000 useable gallons. The non-seismic main condenser hotwell, with a surveilled capacity of 150,000 useable gallons, can be aligned to EFP-2 and EFP-1. Fire Service Water Storage Tanks (FST-1A & FST-1B) can also be aligned to either EFP-1, EFP-2 or EFP-3 with a surveilled capacity of 600,000 useable gallons.

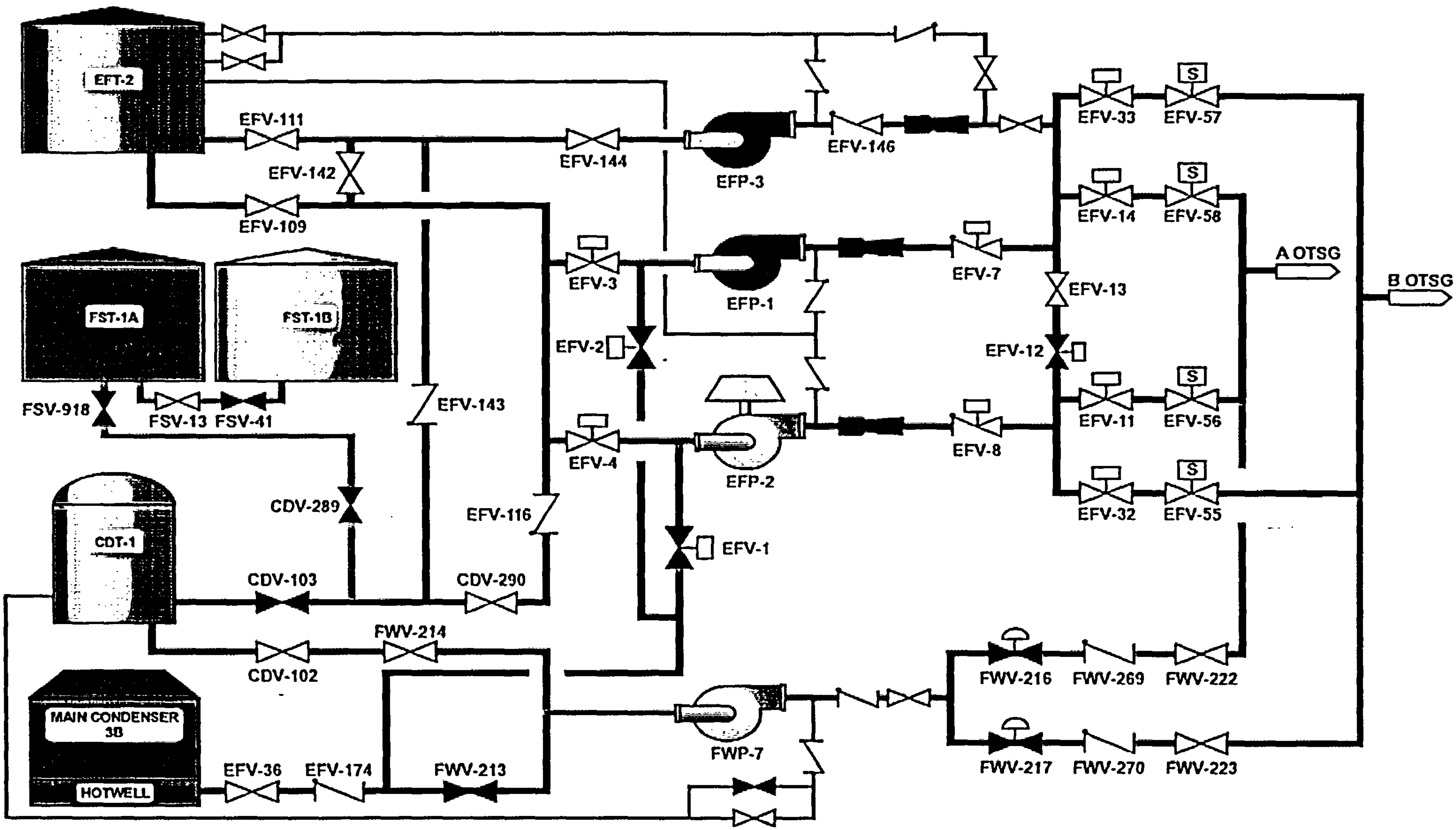

Figure 1 – Emergency and Auxiliary Feedwater

Technical Evaluation

The performance of EFP-3 has recently been evaluated and it was concluded that the pump is fully capable of supporting CR-3 operation. Although EFP-3 has exhibited elevated vibration levels, it is fully capable of performing its safety function. Consistent with the requirements of the Operating and Maintenance Code (OM), Part 6 and the CR-3 In-Service Testing Program EFP-3 is not on increased testing frequency.

During the requested extended time period of 14 days, EFP-2 (redundant "B" train pump) will be available and capable of providing OTSG cooling during emergency conditions. EFP-1 is safety-grade, and, although it is not automatically initiated, it will also be available and capable of OTSG cooling for all design basis events where offsite power remains available or where sufficient emergency diesel generator (EDG) capacity is available. FWP-7 is also capable of providing OTSG cooling for all but the most limiting design basis events if offsite power is available or its non-safety diesel backup power is available. CR-3 Emergency Operating Procedures incorporate the use of EFP-1 and FWP-7 if EFP-2 and EFP-3 are not available.

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. Other compensatory actions that may be implemented include: use of pre-job briefings and periodic operator walkdowns to assess the status of risk sensitive equipment in the redundant train, use of operator walkdowns to assess and limit transient combustibles in risk significant fire areas identified in Attachment E of this submittal, and no elective maintenance to be scheduled in the switchyard that would challenge the availability of offsite power to the ES buses. These actions are more fully described in subsequent sections.

Deletion of the 10-day Completion Time is consistent with TSTF-439, Revision 1 which deletes the second Completion Time from several LCOs and corresponding Bases including ITS 3.7.5, Required Action B.1, 10 day Completion Time. TSTF-439 discusses the acceptability of deleting the 10 day completion Time is based on the inclusion of structures, systems and components (SSCs) in the scope of the Maintenance Rule which would identify if continuous multiple entries into the ACTIONS of the ITS results in unacceptable unavailability of these SSCs.

Risk Evaluation

Attachment E provides the calculation performed to assess the risk associated with increasing the ITS Completion Time to perform repairs to EFP-3. The calculation includes the risk associated with having an Emergency Feedwater pump out-of-service for 14 days using the current CR-3 Equipment Out-Of-Service (EOOS) computer model based on the most current plant Probabilistic Safety Analysis (PSA).

The PSA risk associated with the activity to repair the Emergency Feedwater pump supports the one-time extension proposed in this LAR. Assuming EFP-3 is out-of-service for 14 days, the bounding risk for this activity is estimated with a Change in Core Damage Frequency (ΔCDF) of 4.7E-07. This number 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-06 and is considered a very small increase. 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 14-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 current 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 Probabilistic Safety Assessment (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, Chilled Water, Emergency Feedwater System, Emergency Feedwater Initiation and Controls System (EFIC), Auxiliary Feedwater Pump].

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

1. Increase of operator attention to loss/restoration of redundant train Control Complex Chiller [walkdowns of the operable redundant train, pre-job discussion on the impact of losing service water and the potential EFIC control problems if there is a loss of the redundant train Control Complex Chiller].

2. Operator attention to potential use of the Appendix R Chiller, non-safety grade FWP-7 and Standby Diesel Generator (MTDG-1).

3. Periodic operator walkdowns of the redundant train.

4. No elective maintenance to be scheduled in the switchyard that would challenge the availability of offsite power to the ES Buses.

5. No elective maintenance on defense-in-depth EFW and AFW equipment.

6. EFP-3 extended maintenance will not be performed if adverse weather, as defined by Emergency Preparedness procedures, is anticipated.

7. Establishing fire risk reduction actions such as limiting hot or spark producing work and periodic walkdowns to assess and to limit transient combustibles in risk significant areas (See Attachment E).

8. Consideration should be given to Running EOOS before voluntary realignment of major safety components (e.g., swapping Makeup Pumps).

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

Based on the above evaluation, PEF believes that approval of the proposed change to ITS 3.7.5 will pose an insignificant risk to the plant or to the health and safety of the public.

Precedent

There are similarities between this submittal and the request made by Exelon Generation to the NRC dated June 11, 2003, Request for a License Amendment for a One-Time Extension of the Essential Service Water Train Completion Time. The submittal was applicable to the Braidwood Station, Units 1 and 2 and to the Byron Station, Units 1 and 2. The NRC had previously approved a similar change for the Donald C. Cook Nuclear Plant in Amendments No. 270 and No. 251 for Units 1 and 2, respectively, issued September 9, 2002.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #281, Revision 0

Regulatory Analysis

No Significant Hazards Consideration Determination
Applicable Regulatory Requirements
Environmental Impact Evaluation

No Significant Hazards Consideration Determination

License Amendment Request (LAR) #281, Revision 0, proposed changes include a change to Improved Technical Specifications (ITS) 3.7.5 to allow a one-time increase in the Completion Time for restoring an inoperable Emergency Feedwater (EFW) System train to Operable status online.

This LAR proposes to extend the Completion Time of ITS 3.7.5, Required Action B.1 from 72 hours to 14 days. 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 restoring an inoperable Emergency Feedwater System train to Operable status. The Emergency Feedwater System is designed to provide cooling for components essential to the mitigation of plant transients and accidents. The system is not an initiator of design basis accidents. During the requested extended time period of 14 days, the redundant Emergency Feedwater Pump (EFP) will be available and capable of providing cooling to the Once-Through Steam Generators (OTSGs) during emergency conditions. In addition, a safety-grade motor driven pump (EFP-1) is available for manual initiation and is capable of providing adequate EFW flow for OTSG cooling during all design basis events. EFP-1 is also capable of being supplied by the "A" train emergency diesel generator if sufficient electrical loading capacity is available during a loss of offsite power condition. Although Feedwater (FW) pump FWP-7 is non-safety related and its motor is non-seismic, it will also be available and capable of providing OTSG cooling during all but the most limiting design basis events. FWP-7 also has a non-safety diesel backup power supply in the event normal power is not available.

A Probabilistic Safety Assessment (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 and Large Early Release Frequency, the value of these increases are considered as very small 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 restoring an inoperable Emergency Feedwater System train to Operable status.

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 restoring an inoperable Emergency Feedwater System train to Operable status. The proposed change will allow online alignment of one of the Emergency Feedwater pumps to improve its reliability, 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 an Emergency Feedwater 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. Other compensatory actions that may be implemented include: use of pre-job briefings and periodic operator walkdowns to assess the status of risk sensitive equipment in the redundant train, use of operator walkdowns to assess and limit transient combustibles in risk significant fire areas, and no elective maintenance to be scheduled in the switchyard that would challenge the availability of offsite power to the ES buses.

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 and Large Early Release Frequency, the value of these increases are considered as very small 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 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 ITS 3.7.5 which 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 ITS 3.7.5.

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 #281, 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

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ATTACHMENT C

LICENSE AMENDMENT REQUEST #281, REVISION 0

Proposed Revised Improved Technical Specifications Pages

Strikeout/Shadowed Format

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

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to the turbine driven EFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One EFW train inoperable for reasons other than Condition A.	B.1 Restore EFW train to OPERABLE status.	72 hours * <u>AND</u> 10 days from discovery of failure to meet the LCO*

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

ACTIONS (continued)

B.1

If one of the EFW trains is inoperable, action must be taken to restore the train to OPERABLE status within 72* hours. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low robability of a DBA occurring during this time period. This condition includes the loss of two steam supply lines to the turbine driven EFW pump.

The 10 day* Completion Time for Required Action B.1 established a limit on the maximum time allowed for any combination of Conditions to be entered during any continuous failure to meet this LCO. The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The '<u>AND</u>' connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

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

C.1 and C.2

If Required Action A.1 or Required Action B.1 cannot be completed 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 4 within 12 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.

D.1

With both EFW trains inoperable, the plant is in a seriously degraded condition with no safety related means for conducting a cooldown. In such a condition, plant operation should not be perturbed by a forced action, including a power change, that might result in a trip. For this reason, the Technical Specifications do not mandate a plant shutdown. Rather the ACTIONS allow the plant to dictate the most prudent course of action (including plant shutdown) for the situation. The seriousness of this condition requires that action be initiated immediately to restore at least one EFW train to OPERABLE status.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT D

LICENSE AMENDMENT REQUEST #281, REVISION 0

Proposed Revised Improved Technical Specifications Pages

Revision Bar Format

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to the turbine driven EFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One EFW train inoperable for reasons other than Condition A.	B.1 Restore EFW train to OPERABLE status.	72 hours * AND 10 days from discovery of failure to meet the LCO*

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

BASES

ACTIONS (continued)

B.1

If one of the EFW trains is inoperable, action must be taken to restore the train to OPERABLE status within 72* hours. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a DBA occurring during this time period. This condition includes the loss of two steam supply lines to the turbine driven EFW pump.

The 10 day* Completion Time for Required Action B.1 established a limit on the maximum time allowed for any combination of Conditions to be entered during any continuous failure to meet this LCO. The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The 'AND' connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

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

C.1 and C.2

If Required Action A.1 or Required Action B.1 cannot be completed 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 4 within 12 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.

D.1

With both EFW trains inoperable, the plant is in a seriously degraded condition with no safety related means for conducting a cooldown. In such a condition, plant operation should not be perturbed by a forced action, including a power change, that might result in a trip. For this reason, the Technical Specifications do not mandate a plant shutdown. Rather the ACTIONS allow the plant to dictate the most prudent course of action (including plant shutdown) for the situation. The seriousness of this condition requires that action be initiated immediately to restore at least one EFW train to OPERABLE status.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT E

LICENSE AMENDMENT REQUEST #281, REVISION 0

PSA Risk Assessment of EFP-3 Extended AOT

PSA Risk Assessment of EFP-3 Extended AOT

1.0 Purpose

This analysis calculates the quantitative impact of a proposed one-time risk informed Technical Specification change in the Allowable Outage Time (AOT) for Emergency Feewater Pump (EFP-3) from 72 hours to 14 days (336 hours) in order to perform a special maintenance activity online. EFP-3 is the "A" train Emergency Feedwater (EFW) pump. This is a plant-specific evaluation using the Crystal River Unit 3 (CR-3) Probabilistic Safety Analysis (PSA) model for online operation (Reference 1). The assessment follows the guidance set forth in NRC Regulatory Guide 1.174 (Reference 2). It evaluates the changes in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), and provides suggested compensatory actions to minimize risk impacts of the proposed maintenance.

2.0 References

1. CR-3 calculation P-02-0001, Rev.0, "CR3 PSA Model of Record," November 2003
2. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"
3. CR-3 Individual Plant Evaluation of External Events (IPEEE), Rev.1, March 1997
4. CR-3 Improved Technical Specifications (ITS)

3.0 Design Inputs

The primary input for this analysis is the CR3 PSA Model of Record (MOR) (Reference 1).

The estimated time required to perform the proposed maintenance activity is 7 days based on input from engineering. Based on Institute of Nuclear Operations (INPO) recommended practice, maintenance activities are not normally scheduled for more than half of the AOT. Therefore, this analysis will use twice the estimated schedule time as the expected AOT or 14 days.

4.0 Assumptions

4.1 General

1. The only ITS change will be the AOT for EFP-3. All other limiting conditions will remain unchanged.
2. External events can be evaluated qualitatively.

4.2 At Power default configuration baseline (same as MOR)

1. The plant is operating at 100% power
2. Makeup Pump (MUP-1B) is running to provide normal RCS make-up
3. MUP-1B is Engineered Safeguards (ES) selected for High Pressure Injection (HPI) and powered from the "A" bus
4. MUP-1C is ES selected for HPI and aligned to the Decay Heat Closed Cycle Cooling (DC) system for cooling

5. MUP-1A is not ES selected and aligned to the Nuclear Services Closed Cycle Cooling Water (SW) system for cooling
6. All MU system unavailability is applied to unselected MUP (MUP-1A)
7. Nuclear Services Seawater System Pump (RWP-1) and Nuclear Services Closed Cycle Cooling Water Pump (SWP-1C) are operating to provide normal SW cooling.
8. "A" train Heat Ventilation and Air Conditioning (HVAC) equipment is running (CHHE-1A, CHP-1A, AHF-17A, AHF-19A, AHF-54A)
9. ES 4160V "A" bus is powered from the offsite power transformer (OPT).
10. ES 4160V "B" bus is powered from the backup ES transformer (BEST).

5.0 Calculation/Analysis Details

5.1 Software

The following table contains a listing of the software used to perform this analysis.

Table 1, PRA Software/Tools

Name	Version	Description
CAFTA	4.0b	Manages PRA fault trees, databases, and results. It can also be used as a front end for quantification.
EOOS	3.1x39	Front end for performing PRA quantification and importance/risk analyses.
FORTE	2.2g	Engine for performing PRA quantification.
QRECOVER	2.0	Engine for applying rule based recoveries to cutsets.

5.2 Baseline CDF and LERF

The baseline CDF and LERF for this analysis is taken from Reference 1 as 7.49E-06/year and 3.42E-07/year. They are based on solving the model using Equipment Out-Of-Service (EOOS) as the front end and a truncation of 1E-10, using default alignments.

5.3 Annual increase in CDF with EFP-3 maintenance increased by 14 days

If the 14 day AOT is used, the annual unavailability of EFP-3 could be increased by up to 336 hours. The current unavailability used in the PRA is 1.0E-02. Based on one year, this translates to:

*1.0E-02 * 8760 hours/year = 87.6 hours/year*

Increasing this by 336 hours and converting back to a probability gives:

(87.6 hours + 336 hours) / 8760 hours/year = 4.84E-02

Substituting this value back into the PRA model and solving for CDF gives 7.96E-06/year. This represents a potential increase in CDF (dCDF) for the year of:

dCDF = 7.96E-06/yr – 7.49E-06/yr = 4.70E-07/yr

Per Reg. Guide 1.174, changes in CDF less than 1E-6 are considered very small.

5.4 Instantaneous CDF (ICDF) and with EFP-3 train out of service

Another approach to evaluating the risk of equipment out of service (OOS) is to assess the peak CDF when the component is OOS. Solving the model with the EFP-3 maintenance event set to 1.0 results in an ICDF of 2.18E-05/year.

This value can also be used to develop a Risk Achievement Worth (RAW) for EFP-3 as follows:

$$RAW = ICDF / baseCDF = 2.18E\text{-}05/yr / 7.49E\text{-}06/yr = 2.91$$

5.5 Incremental Core Damage Probability (ICDP)

ICDP is equal to the integration of the change in risk over time. For this assessment it can be determined as the increase in risk (dICDF) when EFP-3 is OOS times the duration of the AOT.

$$dICDF = ICDF - baseCDF = 2.18E\text{-}05/yr - 7.49E\text{-}06/yr = 1.43E\text{-}05/yr$$

$$ICDP = dICDF * AOT = 1.43E\text{-}05/yr * 336\ hours * (1yr/8760\ hours) = \mathbf{5.48E\text{-}07}$$

5.6 Impact on LERF

Similar to impact on CDF, the impact to LERF is another metric to be evaluated. The baseline LERF from Reference 1 is 3.42E-07/yr. When the maintenance probability for EFP-3 is increased to 4.84E-02, the annual LERF increases to 3.47E-07/yr. When EFP-3 is OOS, the instantaneous LERF value increases to 5.08E-07/yr. Therefore:

$$dLERF = 3.47E\text{-}07/yr - 3.42E\text{-}07/yr = \mathbf{5.00E\text{-}09/yr}$$

and

$$dILERF = ILERF - baseLERF = 5.08E\text{-}07/yr - 3.42E\text{-}07/yr = 1.66E\text{-}07/yr$$

$$ICLERP = dILERF * AOT = 1.66E\text{-}07/yr * 336\ hours * (1yr/8760\ hours) = \mathbf{6.37E\text{-}09}$$

Per Reg Guide 1.174, changes in LERF less than 1E-7 are considered very small.

5.7 Sensitivities and Zero Maintenance Baseline

In addition to the risk of EFP-3 alone, consideration must be given to other equipment outages that should be prevented while EFP-3 is in the extended AOT. Removing certain equipment from service, such as the other train of EFW (EFP-2), is already prohibited by Technical Specifications which will be followed. Other equipment will be managed based on risk and current operating practice.

This will be performed similar to the normal risk management practice at CR-3, using the A4/EOOS model (Reference 2). This model is identical to the model of record except that it is based on a zero maintenance baseline (all maintenance probabilities set to 0.0), and is solved at a higher truncation (1E-08) which allows for quicker solution times. The baseline zero maintenance CDF is 2.68E-06/yr. Although the absolute CDF values are lower, the relative changes and risk insights provided are consistent with solutions at lower truncations. The relative risks are presented as a Risk Achievement Worth (RAW) value where:

Risk = RAW = (ICDF for a given configuration) / (zero maintenance CDF)

Current risk management practices would require enhanced communication and practical compensatory actions for configurations with a risk > 10. Risks > 45 would not be scheduled under normal circumstances. In addition, the ICDP for each configuration would be evaluated based on the scheduled OOS durations. For the proposed activity, configurations with a risk over 10.0 should be avoided while in the extended AOT period. In addition, EFP-1 should be kept available for defense-in-depth.

Table 2, Relative Risks for Equipment Out-of-Service

Component (w / EFP-3)	Description	Risk Alone	Risk w/EFP-3
CHP-1A	"A" train Control Complex HVAC (EFIC support system)	1.0	11.4
CHP-1B	"B" train control complex HVAC (EFIC support system)	2.5	5.4
CHP-2	App. R HVAC (EFIC support system defense in depth, "B" AC power supported)	1.7	5.5
DCP-1A	"A" train Decay Heat Closed Cycle Cooling (DHCCC)	13.3	31.2
DCP-1B	"B" train Decay Heat Closed Cycle Cooling (DHCCC)	16.7	45.1
DHP-1A	"A" train Decay Heat Removal	13.2	30.9
DHP-1B	"B" train Decay Heat Removal	13.5	36.9
EFP-1	"A" train EFW Defense in depth (manually actuated, AC driven)	1.0	3.4
EFP-2	"B" train EFW (steam driven)	1.8	41.0
EGDG-1A	"A" train Emergency AC power	2.5	13.1
EGDG-1B	"B" train Emergency AC power	5.0	18.2
FWP-7	EFW defense in depth (non-safety AC diesel generator backed)	4.6	20.0
MTDG-1	FWP-7 non-safety diesel generator	1.0	2.9
MUP-1A	"A" train HPI or standby	1.0	2.6
MUP-1B	"A/B" swing HPI or standby	1.0	2.4
MUP-1C	"B" train HPI or standby	3.2	8.7
RWP-1	Non-safety normal duty sea-water cooling to NSCCC	1.0	2.5
RWP-2A	"A" train safety related sea-water cooling to NSCCC	5.4	11.5
RWP-2B	"B" train safety related sea-water cooling to NSCCC	1.6	3.6
RWP-3A	"A" train safety related sea-water cooling to DHCCC	13.3	31.2
RWP-3B	"B" train safety related sea-water cooling to DHCCC	16.7	41.5
SWP-1A	"A" train Nuclear Services Closed Cycle Cooling (NSCCC)	5.4	11.5
SWP-1B	"B" train NSCCC	1.5	3.5
SWP-1C	Non-safety normal duty NSCCC	1.1	2.5

5.8 External Events (Fire)

The only external events analysis for CR-3 is contained in the IPEEE submitted in March 1997. This was before EFP-3 was installed at CR-3, therefore there are no direct insights available for EFP-3. FWP-7 is included in the IPEEE however MTDG-1 and control room operation is not. While EFP-3 is in maintenance, EFP-2 will be in standby and controlled by EFIC. EFP-1 can be manually started with flow controlled by EFIC. FWP-7 can be manually started and controlled from the control room. Also, the flowpath from FWP-7 is separate from the EFIC controlled EFW flowpath.

In order to preserve EFP-2 and EFW defense-in-depth, fire zones identified in the IPEEE which can impact these pumps should be monitored more closely to reduce the chance of fire in these areas. The following list identifies the zones which should be considered. This table does not consider fire wrap or suppression capabilities, but can be used as input for further engineering evaluation for fire risk reduction opportunities.

Fire Zone	ASV-5	EFP-1	EFV-55	EFV-56	EFV-57	EFV-58	FWP-7
AB-119-6A		x					
AB-119-6J	x						
AB-95-3B	x	x	x				
CC-108-102	x	x	x	x	x	x	
CC-108-103	x	x	x	x	x	x	
CC-108-104		x			x	x	
CC-108-105	x	x	x	x	x	x	
CC-108-106		x					
CC-108-107	x		x	x			
CC-108-108		x			x	x	
CC-108-109		x					
CC-108-110		x					
CC-124-111		x	x	x	x	x	
CC-124-112		x			x	x	
CC-124-115	x		x	x			
CC-124-116	x		x	x			
CC-124-117		x	x	x			
CC-134-118A	x	x	x	x	x	x	x
CC-145-118B		x	x	x	x	x	x
CC-1-8-107							
IB-119-201A	x					x	
IB-119-201B	x	x	x	x			
IB-95-200B		x					
IB-95-200C	x	x	x	x	x	x	
TB-119-400E							x
TB-119-403							x
TB-95-400A							x
TB-95-401							x

6.0 Results / Conclusions

Based on the analyses performed, the quantitative data associated with a one-time extended AOT of 14 days show the risk to be acceptable per the current regulatory guidance.

The delta CDF of 4.7E-07/yr is below the Reg. Guide threshold of 1E-06 and is considered very low risk. Also, the delta LERF of 5.0E-09/yr is below the Reg. Guide threshold of 1E-07 and is considered very low risk.

This assessment is based on appropriate use of risk management actions including limiting work on defense-in-depth equipment and other risk significant systems. Risk assessment should be performed prior to and at the time proposed extended AOT is entered.