Progress Energy

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

Ref: 10 CFR 50.90

June 9, 2005
3F0605-01

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

Subject: Crystal River Unit 3 – License Amendment Request #289, Revision 1
Revised Improved Technical Specifications (ITS) 3.5.2, Emergency Core Cooling Systems (ECCS) - Operating, 3.6.6, Reactor Building Spray and Containment Cooling Systems, 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System and 3.7.10, Decay Heat Seawater System (TAC No. MC5631)

Dear Sir:

Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF), is hereby providing Revision 1 to License Amendment Request (LAR) #289. This revision to LAR #289 is necessary to update the probabilistic safety assessment that supports the acceptability of the changes proposed in LAR #289. LAR #289, Revision 0, Reference 1 (References are provided on Page 6 of Attachment A), was submitted on January 13, 2005. LAR #289, Revision 0, requested a one-time change to the Crystal River Unit 3 (CR-3) Facility Operating License in accordance with 10 CFR 50.90. LAR #289 proposed a one-time 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.

Attachment A of this submittal provides additional information and an updated evaluation describing changes to plant conditions that have required a revision to the Risk Assessment previously provided in Attachment E of Reference 1. The specific plant condition is a change in the normal position of the Power Operated Relief Valve (PORV) Block Valve (RCV-11) to be closed. This has been required in order to isolate a Reactor Coolant System (RCS) to Reactor Building atmosphere leak (approximately 2.5 gallons per minute) which was discovered on March 3, 2005, following the quarterly stroke test of RCV-11. In accordance with the evaluation performed for Administrative Instruction AI-506, "Operational Decision Making," CR-3 will be operating with the RCV-10/11 flow path closed during normal operations until Refueling Outage 14 scheduled for Fall 2005. RCV-11 will be opened during certain Emergency Operating Procedure/Abnormal Procedure (EOP/AP) events to allow usage of the PORV during these events.

Attachment B provides Calculation P-05-0001, Revision 1, "PSA Risk Assessment of RWP-3B Extended AOT," that has been revised to evaluate the risk impacts of operating with the PORV (RCV-10) and Block Valve (RCV-11) in this configuration during the proposed extended Allowed Outage Time (AOT) for refurbishing RWP-3B.

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

A001

No changes have been made to the No Significant Hazards Determination provided in Attachment B of Reference 1. No changes have been made to the proposed Improved Technical Specification pages provided in Attachments C and D of Reference 1.

Revision 1 to LAR #289 makes no changes to References 2, 3 and 4. Reference 2 provided additional information concerning fire detection, suppression, ignition sources, and combustible loads. The submittal included additional compensatory actions for risk significant fire zones and operational controls to be taken for the duration of the RWP-3B maintenance activity. Reference 3 provided an assessment of LAR #289, Revision 0, on Section 9.7 of the Security Plan. Reference 4 provided CR-3 responses to an NRC Request for Additional Information (RAI).

CR-3 will implement the commitments stated in References 1, 2, and 4 during the proposed one-time extended AOT. No new commitments are included in this submittal.

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

PEF is respectfully re-stating the request for an approval date for LAR #289, Revision 1 of August 1, 2005. This approval date was previously requested in Reference 4.

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, Deterministic Considerations, Impact of RCV-11 Closure, Roles of PORV Use in EOP/AP, Summary of EOP/AP Changes and Conclusion, Risk Considerations and Conclusion, and References
B. PSA Risk Assessment of RWP-3B 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 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
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 9th day of June, 2005, by Dale E. Young.

Signature of Notary Public
State of Florida

Ellen Deppolder
My Commission DD040101
Expires July 08, 2005

(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 1

Background, Deterministic Considerations, Impact of RCV-11 Closure, Roles of PORV Use in EOP/AP, Summary of EOP/AP Changes and Conclusion, Risk Considerations and Conclusion, and References

BACKGROUND

The information which was provided in Attachment A, Reference 1 (References are provided on Page 6 of this Attachment) is hereby being updated with additional information describing changes to plant conditions that have also required a revision to the Risk Assessment previously provided in Attachment E of Reference 1.

After starting up from Refueling Outage 13 (13R), a leakage path developed that allowed the Reactor Coolant System (RCS) to leak through the pressurizer Power Operated Relief Valve (PORV), RCV-10, assembly into the Reactor Coolant Drain Tank (RCDT). RCS leakage into the RCDT was indicated by the elevated RCV-10 tail pipe temperatures (from 200°F to 212°F). Other indications of leakage through the PORV such as pressurizer heater demand, RCDT level and pressure change, and Waste Gas Hydrogen Concentration were also monitored. These indicated that the PORV leak contributed a small fraction of the RCS leakage as calculated in Surveillance Procedure SP-317, "RC System Water Inventory Balance." The leakage rate past the PORV has not measurably increased with time.

On March 3, 2005, following the quarterly stroke test of the PORV Block Valve (RCV-11), an RCS to Reactor Building atmosphere leak (approximately 2.5 gallons per minute) was discovered. The RCS leak is located downstream of RCV-11 which is a normally open valve that was closed to isolate the leak. Closure of the Block Valve on March 3, 2005, placed the valve in its required safety position. The stroke test satisfied its surveillance requirements for the first quarter of 2005. An evaluation was performed in accordance with Administrative Instruction AI-506, "Operational Decision Making," and a decision was made to operate with the RCV-10/11 flow path closed during normal operations until Refueling Outage 14 scheduled for Fall 2005. RCV-11 will be opened during certain Emergency Operating Procedure/Abnormal Procedure (EOP/AP) events to allow usage of the PORV during these events.

The surveillance requirement of Improved Technical Specification (ITS) Surveillance Requirement (SR) 3.4.10.1 for the second quarter 2005 has not been performed and therefore, SR 3.4.10.1 has not been met and the block valve has been declared inoperable. The failure to perform the quarterly surveillance has not reduced the reliability of the Block Valve to perform its design function of isolating a stuck open PORV. The Block Valve has shown a strong history of being reliable. Maintaining the Block Valve closed is consistent with the requirements of ITS 3.4.10, Condition B.

The leak path past the PORV and into its tail pipe remains isolated from the RCS by the closure of the Block Valve. The leak path directly to the Reactor Building (RB) atmosphere is also isolated by the closure of the Block Valve.

The risk impacts of the Block Valve closure have been evaluated in revised Calculation P-05-0001, Revision 1 (Attachment B of this submittal).

An update to the License Amendment Request LAR #289 technical evaluation discussing the impact of the Block Valve closure is provided below. The evaluation concludes that there is no adverse impact on the requested one-time allowed Outage Time (AOT) extension needed to refurbish RWP-3B. The conclusion is based on both deterministic and risk informed considerations.

DETERMINISTIC CONSIDERATIONS

Impact of the Block Valve Closure

The PORV is a pilot operated, relief valve with both automatic and manual functions which provide the ability to reduce RCS pressure. The PORV was installed as a non-safety related component and is not credited in the CR-3 design basis for any design basis Final Safety Analysis Report (FSAR) chapter 14, accident.

The safety related, motor operated, Block Valve, which can be operated from the control room, is installed upstream of the PORV and can be used to isolate the PORV flow path if necessary. The CR-3 Technical Specifications allow the plant to operate indefinitely with the PORV flow path isolated.

As part of the AI-506 evaluation, the Block Valve qualification has been reviewed and evaluated to assess the effects of a small steam leak condition which occurred during normal plant operations, on March 3, 2005 for approximately 2 hours but has since been stopped by closing the Block Valve. The Block Valve has been qualified by test for accident temperature, pressure, humidity and radiological conditions and by test supplemented with analysis for the Reactor peak temperature and associated chemical spray associated with a Loss-of Coolant Accident (LOCA) conditions. Therefore, the small leak is considered to have negligible impact on this motor operated valves' ability to perform its design safety function and that it will continue to operate for the required CR-3 post accident operating time of six (6) months even after exposure to this small steam leak.

The evaluation also documents the qualification status and amount of service life of affected components if the steam leak would happen to re-occur (by opening the valve) anytime within the next 8 months until the leak is either repaired or the valve is replaced. In order to determine if this abnormal environmental condition would affect the operator's ability to cycle the valve, the evaluation focused in determining if any of the motor operator non-metallic (age sensitive) materials will degrade significantly as a result of a postulated steam leak.

The evaluation concluded that the components associated with the Block Valve limit switch and motor components have a minimum service life of 7.5 months under conditions that bound the March 3, 2005 leak (which is currently isolated). This service life exceeds the time remaining in the current operating cycle. Therefore, the Block Valve is expected to perform as designed upon demand during normal or beyond design basis scenarios. Based on the results of the evaluation, the subject valve actuator with motor is not susceptible to significant degradation by exposure to a Boron chemical spray solution, moisture intrusion or temperature (aging) conditions which would prevent the Block Valve from cycling or performing its intended design safety function.

Roles of PORV Use in EOP/AP

The roles of the PORV in Emergency Operating Procedures/ Abnormal Procedures (EOP/AP) beyond design basis scenarios are discussed below:

1. Core Cooling

   High Pressure Injection (HPI)-code safety cooling is the credited method for core cooling instead of HPI-PORV cooling. This is described in FSAR 4.2.4.2.3. Although the automatic function of the PORV is disabled in the current configuration, the PORV is available for use as part of the mitigation strategy for Core Cooling to avoid challenging the RCS Code Safety Valves.

2. RCS blowdown when HPI is ineffective

   For situations where HPI is not sufficient, the PORV can be used to depressurize the RCS to initiate core flood injection, and then eventually, Low Pressure Injection (LPI) cooling to ensure adequate inventory is in the RCS. Although the automatic function of the PORV is disabled, this scenario does not depend on its automatic functionality.

3. Pressure Control

   To ensure that the reactor vessel's Nil Ductility Temperature (NDT) limits are not challenged during a pressurized cooldown, the PORV is used to control pressure. The PORV also minimizes challenges to the pressurizer safety valves. However, other means of RCS pressure control could be used for this event.

4. Tube Rupture Alternate Control Criteria (TRACC)

   TRACC is used to determine when the affected steam generator(s) would need to be isolated (feedwater and steam) to terminate any offsite releases. The PORV, along with pressurizer high point vents and high pressure auxiliary spray, can be used to depressurize the RCS when offsite power is not available. Minimizing the pressure differential between the RCS and secondary side will reduce the break flow rate. Although the automatic function of the PORV is disabled, the PORV is available and could be used to limit the dose consequence of this event.

5. Low Temperature Overpressure Protection (LTOP)

   The PORV is used as an LTOP device, which mitigates the impact of inadvertent pressurizations at low pressure/temperature conditions during normal operations. However, meeting the requirements for LTOP is not needed during accident conditions. Once plant conditions are stabilized, LTOP is established through the normal operating procedures or Technical Support Center (TSC) recovery actions based on the event.

6. Others of less significance

   There are other steps in the EOPs which mention the PORV, but these steps are typically informational or verification steps.

Continued operation until Refueling Outage 14 with the PORV block valve closed but available has been evaluated based on the guidance of procedure AI-506, "Operational Decision Making." This option required the implementation of compensatory measures including EOP changes to support the current plant condition with RCV-10/11 inoperable but available. Other changes included the development of a new procedure for managing an inadvertent Engineered Safeguards (ES) actuation (AP-340), simulator modeling and licensed operator training.

Summary of EOP/AP Changes

The affected EOPs and APs have been revised to ensure a PORV flow path is available to mitigate an event for which it is desirable to use the PORV. The automatic function of the PORV will not be relied upon. The EOP/AP changes are summarized below:

- Revise EOP/AP Details to ensure Block Valve is open prior to cycling the PORV.
- Redefine the order of preference when using the PORV as an option for RCS depressurization. The preferred order will be the Normal Pressurizer Spray if the RCPs are operating, High Pressure Auxiliary Spray (if available), Pressurizer High Point Vent path and then the PORV.
- Not relying on PORV automatic function for LTOP during accident conditions when the Block Valve is not operable.

EOPs/APs affected by the above changes are:

EOP-4, Inadequate Heat Transfer
EOP-5, Excessive Heat Transfer
EOP-6, Steam Generator Tube Rupture
EOP-7, Inadequate Core cooling
EOP-8, Loss-of-Coolant Accident (LOCA) Cooldown
EOP-9, Natural Circulation Cooldown
EOP-14, Emergency Operating Procedure Enclosures
AP-404, Loss of Decay Heat Removal
AP-490, Reactor Coolant System Boration
AP-520, Loss of RCS Coolant or Pressure
AP-545, Plant Runback

Conclusion

The changes made to the EOPs/APs ensure the PORV flow path is available to mitigate an event when it is desirable to use the PORV. The EOP/AP instructions are not being changed except for the order that the PORV might be used based on the effectiveness of other methods (i.e., Normal Pressurizer Spray, High Pressure Auxiliary Spray or Pressurizer High Point Vent path). Not performing the Block Valve quarterly surveillance and the existence of a leak downstream of the Block Valve seat does not significantly affect the reliability of the PORV or the Block Valve.

RISK CONSIDERATIONS

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 risk due to internal events for this activity is estimated with a Change in Core Damage Frequency (ΔCDF) of 4.0E-07/yr. This is considered a very small increase per Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." A sensitivity based on fire risk adds about 2.72E-06/yr. The total risk is below the Regulatory Guide (RG) 1.174, limit of 1E-05 and is considered a 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 assessment also addresses Incremental Conditional Core Damage Probability (ICCDP) values for various plant configurations, including performing the activity with the Block Valve closed. The results indicate acceptable risks given appropriate risk management actions. 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. The calculated internal events ICCDP of 1.18E-06 is greater than that generally accepted for permanent AOT changes per RG 1.177. However, it is well within acceptable limits for performing online maintenance within the scope of maintenance rule guidance, and is reasonable for a one time AOT extension.

Model Changes to Assess the Current Configuration of the PORV and Block Valve

In order to properly assess the actual configuration of the PORV and Block Valve, the CR-3 PRA model was revised. In addition to adding an alignment flag for closing the Block Valve, operator actions were included to control inadvertent HPI actuations which can challenge the Safety Relief Valves (SRVs). These actions are backed up by a new procedure (AP-340). Finally, previously identified changes were included which improve the HPI piggy back logic. This change is significant in that it impacts all small LOCA sequences, reducing the baseline CDF.

The model changes were performed using approved Progress Energy procedures. Both the CR-3 Baseline Model of Record and Equipment Out-Of-Service (EOOS)/A4 applications use the same models, and the application software is controlled by corporate Progress Energy procedures. The models and documentation are updated as engineering calculations and electronic files are maintained in a secure location.

Risk Evaluation Conclusion

CR-3 has evaluated the risks associated with an extended AOT to refurbish RWP-3B. The evaluation included operating with the RCV-10/11 flow path closed until Refueling Outage 14 scheduled for Fall 2005. RCV-11 will be opened during certain EOP/AP events to allow usage of the PORV during these events. CR-3 has previously identified the fire and adverse weather events as significant concerns (References 1 and 2). In order to reduce the risk from these events, CR-3 has proposed compensatory measures (References 1, 2 and 4) that maximize the availability of redundant equipment, minimize the potential for ignition and spread of fires, enhance availability of power sources, increase operator awareness and avoid risk due to violent weather through preparation and planning. No changes have been made to the compensatory actions proposed in References 1, 2 and 4. PEF believes that approval of the one-time 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.

Based on the deterministic and PRA insights discussed above, PEF concludes that approval of the proposed one-time ITS change will pose an insignificant increase in risk to the plant or to the health and safety of the public.

References

1. PEF to NRC letter dated January 13, 2005, 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

2. PEF to NRC letter dated February 11, 2005, Crystal River Unit 3 – Supplemental Information Regarding Risk Significant Fire Zones and Fire Zone Specific Compensatory Actions for 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

3. PEF to NRC Letter dated April 13, 2005, Crystal River Unit 3 – Supplemental Information Regarding License Amendment Request #289, Revision 0, Revised Improved Technical Specifications (ITS) 3.5.2, Emergency Core Cooling Systems (ECCS) - Operating, 3.6.6, Reactor Building Spray and Containment Cooling Systems, 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System and 3.7.10, Decay Heat Seawater System (TAC No. MC5631)

4. PEF to NRC Letter dated May 6, 2005, Crystal River Unit 3 – Supplemental Information and Additional Commitment Regarding License Amendment Request #289, Revision 0, Revised Improved Technical Specifications (ITS) 3.5.2, Emergency Core Cooling Systems (ECCS) - Operating, 3.6.6, Reactor Building Spray and Containment Cooling Systems, 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System and 3.7.10, Decay Heat Seawater System (TAC No. MC5631)

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #289, Revision 1

PSA Risk Assessment of RWP-3B Extended AOT

SYSTEM # N/A
CALC. SUB-TYPE N/A
PRIORITY CODE 4
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
1	Signature D. Miskiewicz	Signature Rishel/Billinger	Signature Billinger for JRCaves
	Name David Miskiewicz	Name Robert Rishel	Name John Caves
	Date 6/2/05	Date 6/6/05	Date 6/6/05

(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
1-14	0					
				AMENDMENTS		
				Letter	Rev	Number of Pages

Rev. #	Revision Summary (list of ECs incorporated)
0	Initial issue of calculation
1	Revised calculation to reflect the updated CR3 PRA model and plant configuration changes with the PORV block valve. The format was also revised to address questions from previous review.

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 2	IN	Design input, Reference

Record of Lead Review

Design P-05-0001, **Revision** 1

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**

Lead Reviewer (print)	(sign)	Discipline	Date
Robert Rosher/B. Morgan	Robert Rosher/B. Morgan	PSA	6/6/05

Item No.	Deficiency	Resolution
1)	a. Delete the discussion of fire watches in section 5.2 unless they are planned. b. How did the list of potential fire areas be determined?	The fire watch info has been reviewed and accepted by operations. This information is consistent with rev.0
2)	RG 1.177 is not used in the conclusion to support acceptability. Some discussion of RG 1.177 should be added as well as continuing to implement the existing Risk management actions for Def in depth.	Discussion added in section 6.0
3)	The fire ignitions source frequency for AB-73-5, AC-95-3L are 2.01E-04 vice 2.02E-04. No effect on results. AB-119-6A is 9.73e-05 vice 8.98e-04. AB-119-6E 1.73e-04 vice 9.73e-04	The numbers for 6A,6E, in the calculation are the correct values They each include an additional 8e-4 to account for hydrogen sources in those areas. The H2 sources were added based on an RAI question. Rounding issues account for the other differences depending on whether the IPEEE table or the source documents are used.
4)	I do not understand where the 0.1 came from for CCDP due to RWP-3B. please explain in more detail.	0.1 was chosen as a conservative estimate to use as a sensitivity since the CR3 IPEEE can not be resolved similar to the internal events model. This value has been successfully used in similar applications of this type.
5)	Based upon the Fire risk evaluation, section 5.4 has to state the no transient combustibles will be allowed in the YES rooms, otherwise the fire risk evaluation is not valid or need fire watches.	The last row of the compensatory actions in section 5.4 discuses the use of fire watches to manage the transient combustibles.
6)	No discussion of risk increase due to fire with PORV block closed and RWP-3B OOS.	There is no impact due to the Block valve closure. Ignition frequencies are unchanged and no additional failure modes were created.

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	T.D. SALUTE	[signature]	6/6/05
Operations	Kenneth C. Rass	[signature]	6/6/05
Other Licensing	Loretta V Cecilia	[signature]	6/6/05

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	T.D. SALUTE	[signature]	6/6/05
Comments:			
Operations ☒ Yes ☐ NO	Kenneth C. Rass	[signature]	6/6/05
Comments:			
Licensing	Loretta V Cecilia	[signature]	6/6/05
Comments:			
Other			
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.2, "CR3 PSA Model of Record - MOR_03b"
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
7. NUMARC 93-01, Rev. 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

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.

The normally modeled unavailability for RWP-3B is 0.0086. Based on the one-time AOT of 10 days during the current cycle, this probability can be updated to 0.0223 to determine the delta CDF.

(0.0086) + (10days/2years)*(1year/365days) = 0.0223

Sensitivity cases were also developed based on increasing the probability of loss of offsite power.

The results from all cases considered can be 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 base	CDFi	dCDFi	ICCDP (72h)	ICCDP (7d)	ICCDP (10d)	Days to 1E-06	LERF base	LERFi	dLERFi
	MOR_3b										
1	base (1)	5.40E-06							3.98E-07		
2	base w/RWP-3B maintenance increased (2)		5.80E-06	4.00E-07						3.98E-07	negl
3	zero maintenance (3)	3.23E-06									
4	RWP-3B OOS, MUP-1C (SW)		2.71E-05	2.39E-05	1.96E-07	4.58E-07	6.54E-07	15			
5	RWP-3B OOS, MUP-1A/B ES		2.47E-05	2.15E-05	1.76E-07	4.12E-07	5.88E-07	17			
6	RWP-3B OOS, MUP-1A/B ES, RCV-11 closed (4)		4.64E-05	4.32E-05	3.55E-07	8.28E-07	1.18E-06	8			
7	case 6, w/LOSP x3 (5)		4.82E-05	4.50E-05	3.70E-07	8.62E-07	1.23E-06	8			

Notes	
(1)	Internal events & floods, average maintenance, truncation @ 1E-11
(2)	RWP-3B cycle maintenance unavailability increased by 10 days based on one-time AOT.
(3)	all maintenance events set to zero
(4)	RCV-11 closed but available with manual action for use with PORV as directed by EOPs
(5)	Both LOOP initiators (T3 & t15) increased by a factor of 3

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 values used as baselines when determining dCDF for various cases
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)).
ICLERP (time) –	This is the Incremental Conditional Large Early Release Probability for the case and a specified time period (= $dLERFi_{case}$ * (time/1yr))
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). It is also the threshold recommended by NUMARC 93-01 for implementing risk management actions supporting typical 10CFR50.65 (a)(4) risk assessments.

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 instantaneous CDF due to fire based on this table, with RWP-3B out of service, is 9.92E-05/yr. This translates into a ICCDP associated with RWP-3B AOT and fire of:

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

This value could also be considered as a bounding delta CDF based on a one year exposure.

CR3 does not have a fire PRA model that can be used to quantify the effect of the postulated fire scenario on LERF. However, because the predominant contributor to LERF for CR3 are scenarios based on steam generator tube ruptures (SGTR) or interfacing system LOCAs (ISLOCA), the LERF impact is estimated to be very low. A fire in the "A" 4160V switchgear room would not contribute to an increase in the frequency for either of these initiating events, and the likelihood of a fire occurring in the switchgear room coincident with a SGTR or ISLOCA is also very small. Therefore any increase in LERF will be very small. Given the negligible change in LERF due o the internal events assessment, additional contributions from the fire are expected to support the acceptability of the AOT request per the regulatory guide.

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					
	TRANS-6A-H	1.22E-05					
	TRANS-6A-G	1.22E-05					
	TRANS-6A-F	1.22E-05					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	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					
	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					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	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					
	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					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	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					
	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	4.48E-06					

ZONE	SOURCES IPEEE	IGNF_SOURCE	IGNF_ZONE	IGNF_ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	CAB						
	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					
	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.

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 unavailability on other risk significant 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 be procedure. Additional compensatory actions are provided in Table 5 which can further reduce the risk when practical. Their use should be based on the specific plant configuration during the use of the extended AOT.

The risk metric for this activity is estimated with a delta CDF of 4.0E-07/yr based on internal events (case 2). This is below the RG 1.174 limit of 1E-06 and is considered to be a very small risk. The corresponding delta LERF is below 1E-09/yr and is also considered very low based on the RG 1.174 limit of 1E-07. The risk due to fire was estimated using a sensitivity assessment to get a bounding delta CDF of due to fires of 2.72E-06/yr. Specific compensatory actions are planned to manage and reduce this risk.

The ICCDP for the planned activity is 1.18E-06 (case 6) and considers the plant configuration with RCV-11 closed. This risk is acceptable based on industry guidance (Ref. 7) with proper risk management practices. Planned compensatory actions are expected to reduce this risk. Also, the actual work activity is only scheduled to use half of the requested time, which will reduce these values proportionately. The ICCDP is greater than that generally accepted for permanent AOT changes per RG 1.177, however it is well within acceptable limits for performing online maintenance within the scope of maintenance rule guidance [Ref. 7] and is reasonable for a one time AOT extension.

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. 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.