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10 CFR 50.90

February 25, 2016

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

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Subject: License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

References:

1. TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," dated June 14, 2011.
2. Notice of Availability of the "TSTF-505, Revision 1, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b,'" dated March 15, 2012.

Pursuant to 10 CFR 50.90, Exelon Generation Company, LLC (Exelon) is submitting a request for an amendment to the Technical Specifications (TS) for Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2.

The proposed amendment would modify TS requirements to permit the use of Risk Informed Completion Times in accordance with TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b." The availability of this TS improvement was announced in the Federal Register on March 15, 2012 (77 FR 15399).

Attachment 1 provides a description and assessment of the proposed changes. Attachment 2 provides the existing Technical Specification pages marked up to show the proposed changes. Attachment 3 provides the existing Technical Specification Bases pages marked up to show the proposed changes. Changes to the existing Technical Specification Bases, consistent with the technical and regulatory analyses, will be implemented under the Technical Specification Bases Control Program. They are provided in Attachment 3 for information only. Attachment 4 provides a cross-reference between the improved Standard Technical Specifications included in TSTF-505 and the CCNPP plant-specific TS.

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This amendment request contains one regulatory commitment to complete the installation of modifications necessary to reduce internal fire risk prior to implementation of the Risk Informed Completion Time (RCIT) Program. This commitment is listed in Attachment 5.

Attachment 5 contains security-related information and is requested to be withheld from public disclosure under 10 CFR 2.390.

These proposed changes have been reviewed and approved by the site's Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed amendment by February 25, 2017. The amendment shall be implemented within 180 days following NRC approval, or following completion of the modifications described in Attachment 5, on a per unit basis, whichever is later.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the State of Maryland of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Glenn Stewart at (610) 765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 25th day of February 2016.

Respectfully,

D. P. Helker

David P. Helker
Manager - Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments:

1. Description and Assessment
2. Proposed Technical Specification Changes (Mark-Ups)
3. Proposed Technical Specification Bases Changes (Mark-Ups) (For Information Only)
4. Cross-Reference of TSTF-505 and Calvert Cliffs Nuclear Power Plant Technical Specifications
5. Summary of Regulatory Commitments

Enclosures:

1. List of Revised Required Actions to Corresponding PRA Functions
2. Information Supporting Consistency with Regulatory Guide 1.200, Revision 2
3. Information Supporting Technical Adequacy of PRA Models Without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 2
4. Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models
5. Baseline CDF and LERF
6. Justification of Application of At-Power PRA Models to Shutdown Modes
7. PRA Model Update Process
8. Attributes of the CRMP Model
9. Key Assumptions and Sources of Uncertainty
10. Program Implementation
11. Monitoring Program
12. Risk Management Action Examples

cc: USNRC Region I, Regional Administrator
USNRC Project Manager, CCNPP
USNRC Senior Resident Inspector, CCNPP
S. T. Gray, State of Maryland

ATTACHMENT 1

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

<u>Description and Assessment</u>

1.0 DESCRIPTION

Exelon Generation Company, LLC (Exelon) is submitting a request for an amendment to the Technical Specifications (TS) for Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2.

The proposed amendment would modify the TS requirements related to Completion Times (CTs) for Required Actions to provide the option to calculate a longer, risk-informed CT (RICT). A new program, the Risk-Informed Completion Time Program, is added to TS Section 5 Administrative Controls.

The methodology for using the RICT Program is described in NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, which was approved by the NRC on May 17, 2007. Adherence to NEI 06-09 is required by the RICT Program.

The proposed amendment is consistent with TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b." However, only those Required Actions described in Attachment 4 are proposed to be changed.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Exelon Generation Company, LLC (Exelon) has reviewed the model safety evaluation published on March 15, 2012 as part of the Federal Register Notice for Availability. This review included a review of the NRC staff's evaluation, as well as the supporting information provided to support TSTF-505 and the safety evaluation for NEI 06-09. As described in the subsequent paragraphs, Exelon has concluded that the technical basis presented in the TSTF-505 proposal and the associated model safety evaluation prepared by the NRC staff are applicable to Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2, and support incorporation of this amendment in the CCNPP Technical Specifications (TS).

The traveler and model safety evaluation discusses the applicable regulatory requirements and guidance, including the 10 CFR 50, Appendix A, General Design Criteria (GDC). CCNPP is not licensed to the 10 CFR 50, Appendix A, GDC. CCNPP's Updated Final Safety Analysis Report (UFSAR), Section 1C.0, "AEC Proposed General Design Criteria for Nuclear Power Plants," provides an assessment against the draft GDC published in 1967. A review has determined that the CCNPP plant-specific requirements are sufficiently similar to the Appendix A GDC as related to the proposed changes. Therefore, the proposed changes are applicable to CCNPP.

2.2 Verifications and Regulatory Commitments

In accordance with Section 4.0, Limitations and Conditions, of the safety evaluation for NEI 06-09, the following is provided:

1. Enclosure 1 identifies each of the TS Required Actions to which the RICT Program will apply, with a comparison of the TS functions to the functions modeled in the probabilistic

risk assessment (PRA) of the structures, systems and components (SSCs) subject to those actions.

2. Enclosure 2 provides a discussion of the results of peer reviews and self assessments conducted for the plant-specific PRA models which support the RICT Program, as required by Regulatory Guide (RG) 1.200, Section 4.2.

3. Enclosure 3 is not applicable since each PRA model used for the RICT Program is addressed using a standard endorsed by the Nuclear Regulatory Commission.

4. Enclosure 4 provides appropriate justification for excluding sources of risk not addressed by the PRA models.

5. Enclosure 5 provides the plant-specific baseline core damage frequency (CDF) and large early release frequency (LERF) to confirm that the potential risk increases allowed under the RICT Program are acceptable.

6. Enclosure 6 is not applicable since the RICT Program is not being applied to shutdown modes.

7. Enclosure 7 provides a discussion of the licensee's programs and procedures that assure the PRA models that support the RICT Program are maintained consistent with the as-built, as-operated plant.

8. Enclosure 8 provides a description of how the baseline PRA model, which calculates average annual risk, is evaluated and modified for use in the Configuration Risk Management Program (CRMP) to assess real-time configuration risk, and describes the scope of, and quality controls applied to, the CRMP.

9. Enclosure 9 provides a discussion of how the key assumptions and sources of uncertainty in the PRA models were identified, and how their impact on the RICT Program was assessed and dispositioned.

10. Enclosure 10 provides a description of the implementing programs and procedures regarding the plant staff responsibilities for the RICT Program implementation, including risk management action (RMA) implementation.

11. Enclosure 11 provides a description of the implementation and monitoring program as described in NEI 06-09, Section 2.3.2, Step 7.

12. Enclosure 12 provides a description of the process to identify and provide RMAs.

2.3 Optional Changes and Variations

Exelon is proposing optional changes and variations described below from the TS changes described in TSTF-505, Revision 1, or the applicable parts of the NRC Staff's model safety evaluation published on March 15, 2012.

Note that CCNPP uses different numbering and titles than the improved Standard Technical Specifications (STS) in several instances. These differences are administrative and do not affect the applicability of TSTF-505 to the CCNPP TS. Only TS changes consistent with CCNPP's design and TS are included. Attachment 4 provides specific information.

Attachment 4 is a cross reference that provides a comparison between the NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," Required Actions included in TSTF-505 and the CCNPP Required Actions included in this license amendment request. The attachment includes a summary description of the referenced Required Actions, which is provided for information purposes only and is not intended to be a verbatim description of the Required Action. The cross reference identifies the following:

1. CCNPP Required Actions that have identical numbers to the corresponding NUREG-1432 Required Actions are not deviations from TSTF-505, with the exception of administrative deviations (if any) such as formatting. These deviations are administrative with no impact on the NRC's model safety evaluation published on March 15, 2012 (77 FR 15399).

2. CCNPP Required Actions that have different numbering than the NUREG-1432 Required Actions are an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation published on March 15, 2012 (77 FR 15399).

3. For NUREG-1432 Required Actions that are not contained in the CCNPP TS, the corresponding TSTF-505 mark-ups for the Required Actions are not applicable to CCNPP. This is an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation published on March 15, 2012 (77 FR 15399).

4. The model application provided in TSTF-505 includes an attachment for typed, camera-ready (revised) TS pages reflecting the proposed changes. CCNPP is not including such an attachment due to the number of TS pages included in this submittal that have the potential to be affected by other unrelated license amendment requests. This is an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation published on March 15, 2012 (77 FR 15399).

5. There are several plant-specific LCOs and associated Required Actions for which CCNPP is proposing to apply the RICT Program that are variations from TSTF-505 as identified in Attachment 4 with additional justification provided below:

    - CCNPP TS 3.7.3, Auxiliary Feedwater (AFW) System. The CCNPP plant-specific TS 3.7.3 includes TS 3.7.3.B for one motor-driven AFW pump inoperable and TS 3.7.3.C for two AFW pumps inoperable.

      For CCNPP TS 3.7.3, during conversion to the improved Standard Technical Specifications (ISTS), NUREG-1432, TS 3.7.5, Actions A and E were deleted, new Actions A, B, and C were added, and the remaining Actions from NUREG-1432, TS 3.7.5 were renumbered. The NUREG-1432, TS 3.7.5, is based on plants with one turbine-driven AFW pump and two motor-driven AFW pumps. The above changes to the CCNPP TS 3.7.3 are the result of the unique design of the CCNPP AFW system which contains two turbine-driven pumps and one motor-driven AFW pump. In addition, there is the capacity to cross connect Unit 1 and Unit 2 motor-driven AFW pumps. This design required unique Actions that were contained in the original CCNPP custom TS that were incorporated into CCNPP TS 3.7.3 during the conversion to ISTS, e.g., TS 3.7.3.B and TS 3.7.3.C. However, CCNPP also adopted the NUREG-1432 Actions which require AFW trains to be Operable. The summation of these changes resulted in the current CCNPP TS 3.7.3.

- CCNPP TS 3.7.6, Service Water (SRW) System. The CCNPP plant-specific TS 3.7.6 includes TS 3.7.6.A for one SRW heat exchanger inoperable.

  The SRW system consists of two subsystems that remove heat from various components. Both subsystems are redundant to assure safe operation and shutdown of the plant assuming a single failure. Each SRW subsystem has two plate and frame heat exchangers (PHEs) that operate in parallel. Valves are provided in the SRW system to allow isolation of any selected PHE, while continuing to operate. A single PHE on a subsystem cannot remove the full loss of coolant accident (LOCA) heat load while maintaining SRW temperature within its design limits. However, if one Containment Air Cooler on the affected SRW subsystem is isolated and removed from operation as required by CCNPP TS 3.7.6, Required Action A.1, then the single PHE can remove the remaining accident heat load on the subsystem. Therefore, the SRW subsystem remains operable in this configuration. This plant-specific change was approved via Amendment Nos. 230 (CCNPP Unit 1) and 206 (CCNPP Unit 2) on April 14, 1999 subsequent to the conversion of the CCNPP TS to the ISTS format.

- CCNPP TS 3.7.7, Saltwater System. NUREG-1432 TS 3.7.9 contains requirements for an ultimate heat sink. The CCNPP TS do not contain ultimate heat sink requirements. The Chesapeake Bay, utilizing the saltwater system, is the ultimate heat sink at CCNPP, which provides the cooling medium for the component cooling and service water systems. The saltwater system is a train system similar to the component cooling and service water systems. Because of this, the service water TS were used as a template for the CCNPP saltwater system TS. Therefore, the TSTF-505 TS 3.7.8 (Service Water System) changes applied to the CCNPP service water system TS 3.7.6 are also proposed to be applied to the CCNPP saltwater system TS 3.7.7. The TSTF-505 TS 3.7.9 changes for the ultimate heat sink are not applicable to CCNPP. The CCNPP saltwater system is modeled in the PRA.

- CCNPP TS 3.7.15, Main Feedwater Isolation Valves (MFIVs) (NUREG-1432 TS 3.7.3). The NUREG-1432 MFIV TS was not included in TSTF-505 because the TS LCO Conditions do not include a restoration action for an inoperable MFIV. CCNPP TS 3.7.15 already includes TS 3.7.15.A for one or more MFIVs inoperable which requires restoring one MFIV to an operable status within 72 hours. The MFIVs are modeled in the PRA and credited in the safety analysis to close during a feedwater line break and a steam line break.

The MFIVs at CCNPP do not include bypasses. One MFIV is provided in each of the two main feedwater lines to the steam generators and is required to close within the response time consistent with the safety analyses. These inlet nozzles are separate from the auxiliary feedwater nozzles. These valves are required to be open to support unit operation; the unit cannot be operated with the MFIVs isolated. Therefore, during the conversion to the ISTS, CCNPP TS 3.7.15 revised Required Action A.1 to require that an inoperable MFIV be restored to Operable status rather than be closed. Additionally, the Completion Time for restoring the MFIV to Operable status was chosen to be 72 hours. The 72-hour Completion Time is appropriate due to the unlikely occurrence of an event during this time and since the design at CCNPP also includes a trip of the main feedwater pumps upon receipt of a steam generator isolation signal. Tripping the MFW pumps helps mitigate the events for which the MFIVs are credited.

The main feedwater regulating valves are also automatically closed and main feedwater bypass valves are automatically opened on a turbine trip, reducing main feedwater flow. Although this additional isolation capability is not safety related, this capability is sufficiently reliable to permit a reasonable time for restoration of an inoperable MFIV.

Note: The cross reference does not include existing CCNPP Conditions or Required Actions that are renumbered strictly as a result of adding new Conditions or Required Actions in accordance with TSTF-505. Such changes are administrative (formatting) changes with no impact on the NRC's model safety evaluation published on March 15, 2012 (77 FR 15399). These changes are reflected in the attached TS markups provided in Attachment 2.

Exelon has reviewed these changes and determined that they do not affect the applicability of TSTF-505, Revision 1 to the CCNPP TS.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

Exelon Generation Company, LLC (Exelon) has evaluated the proposed changes to the TS using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2, requests adoption of an approved change to the standard technical specifications (STS) and plant-specific technical specifications (TS), to modify the TS requirements related to Completion Times for Required Actions to provide the option to calculate a longer, risk-informed Completion Time. The allowance is described in a new program in Chapter 5, "Administrative Controls," entitled the "Risk-Informed Completion Time Program."

As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes permit the extension of Completion Times provided the associated risk is assessed and managed in accordance with the NRC approved Risk-Informed Completion Time Program. The proposed changes do not involve a significant increase in the probability of an accident previously evaluated because the changes involve no change to the plant or its modes of operation. The proposed changes do not increase the consequences of an accident because the design-basis mitigation function of the affected systems is not changed and the consequences of an accident during the extended Completion Time are no different from those during the existing Completion Time.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes do not change the design, configuration, or method of operation of the plant. The proposed changes do not involve a physical alteration of the plant (no new or different kind of equipment will be installed).

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes permit the extension of Completion Times provided that risk is assessed and managed in accordance with the NRC approved Risk-Informed Completion Time Program. The proposed changes implement a risk-informed configuration management program to assure that adequate margins of safety are maintained. Application of these new specifications and the configuration management program considers cumulative effects of multiple systems or components being out of service and does so more effectively than the current TS.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Exelon concludes that the proposed changes present 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.

3.2 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL EVALUATION

Exelon has reviewed the environmental evaluation included in the model safety evaluation published on March 15, 2012 (77 FR 15399) as part of the Notice of Availability. Exelon has concluded that the NRC staff findings presented in that evaluation are applicable to Calvert Cliffs Nuclear Power Plant, Units 1 and 2.

The proposed changes would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed changes.

ATTACHMENT 2

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Proposed Technical Specification Changes (Mark-Ups)

1.3 Completion Times

The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

Insert 1

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.3 Completion Times

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. --- NOTE ---- Not applicable when second subsystem intentionally made inoperable. ------------ Two subsystems inoperable.	B.1 Restore subsystems to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

Insert 1, cont.

1.3 Completion Times

When a subsystem is declared inoperable, Condition A is entered. The 7 day Completion Time may be applied as discussed in Example 1.3-2. However, the licensee may elect to apply the Risk Informed Completion Time Program which permits calculation of a Risk Informed Completion Time (RICT) that may be used to complete the Required Action beyond the 7 day Completion Time. The RICT cannot exceed 30 days. After the 7 day Completion Time has expired, the subsystem must be restored to OPERABLE status within the RICT or Condition C must also be entered.

If a second subsystem is declared inoperable, Condition B may also be entered. The Condition is modified by a Note stating it is not applicable if the second subsystem is intentionally made inoperable. The Required Actions of Condition B are not intended for voluntary removal of redundant subsystems from service. The Required Action is only applicable if one subsystem is inoperable for any reason and the second subsystem is found to be inoperable, or if both subsystems are found to be inoperable at the same time. If Condition B is applicable, at least one subsystem must be restored to OPERABLE status within 1 hour or Condition C must also be entered. The licensee may be able to apply a RICT to extend the Completion Time beyond 1 hour if the requirements of the Risk Informed Completion Time Program are met. If two subsystems are inoperable and Condition B is not applicable (i.e., the second subsystem was intentionally made inoperable), LCO 3.0.3 is entered as there is no applicable Condition.

The Risk Informed Completion Time Program requires recalculation of the RICT to reflect changing plant conditions. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

If the 7 day Completion Time clock of Condition A or the 1 hour Completion Time clock of Condition B have expired and subsequent changes in plant condition result in exiting the applicability of the Risk Informed Completion Time Program without restoring the

1.3 Completion Times

inoperable subsystem to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start.

If the RICT expires or is recalculated to be less than the elapsed time since the Condition was entered and the inoperable subsystem has not been restored to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start. If the inoperable subsystems are restored to OPERABLE status after Condition C is entered, Conditions A, B, and C are exited, and therefore, the Required Actions of Condition C may be terminated.

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation-Operating

LCO 3.3.1 Four RPS bistable trip units, associated measurement channels, and applicable automatic bypass removal features for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each RPS Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one RPS bistable trip unit or associated measurement channel inoperable except for Condition C (excore channel not calibrated with incore detectors).	A.1 Place affected bistable trip unit in bypass or trip.	1 hour
	<u>AND</u>	
	A.2.1 Restore affected bistable trip unit and associated measurement channel to OPERABLE status.	48 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>OR</u>	
	A.2.2 Place affected bistable trip unit in trip.	48 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more Functions with two RPS bistable trip units or associated measurement channels inoperable except for Condition C (excore channel not calibrated with incore detectors).	B.1	Place one affected bistable trip unit in bypass and place the other affected bistable trip unit in trip.	1 hour
	AND		
	B.2	Restore one affected bistable trip unit and associated measurement channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
C. One or more Functions with one or more power range excore channels not calibrated with the incore detectors.	C.1	Perform SR 3.3.1.3.	24 hours
	OR		
	C.2	Restrict THERMAL POWER to < 90% RTP.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more Functions with one automatic bypass removal feature inoperable.	D.1	Disable bypass channel.	1 hour
	OR		
	D.2.1	Place affected bistable trip units in bypass or trip.	1 hour
	AND		
	D.2.2.1	Restore automatic bypass removal feature and affected bistable trip unit to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
		OR	
	D.2.2.2	Place affected bistable trip unit in trip.	48 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Functions with two automatic bypass removal feature channels inoperable.	E.1	Disable bypass channels.	1 hour
	OR		
	E.2.1	Place one affected bistable trip unit in bypass and place the other in trip for each affected trip Function. AND	1 hour
	E.2.2	Restore one automatic bypass removal feature and the affected bistable trip unit to OPERABLE status for each affected trip Function.	48 hours OR In accordance with the Risk Informed Completion Time Program

3.3 INSTRUMENTATION

3.3.3 Reactor Protective System (RPS) Logic and Trip Initiation

LCO 3.3.3 Six channels of RPS Matrix Logic, four channels of RPS Trip Path Logic, four channels of reactor trip circuit breakers (RTCBs), and four channels of Manual Trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of being withdrawn.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Matrix Logic channel inoperable.	A.1 Restore Matrix Logic channel to OPERABLE status.	48 hours OR in accordance with the Risk Informed Completion Time Program
B. One channel of Manual Trip, RTCBs, or Trip Path Logic inoperable in MODE 1 or 2.	B.1 Open the affected RTCBs.	1 hour
C. One channel of Manual Trip, RTCBs, or Trip Path Logic inoperable in MODE 3, 4, or 5.	C.1 Open all RTCBs.	48 hours

3.3 INSTRUMENTATION

3.3.4 Engineered Safety Features Actuation System (ESFAS) Instrumentation

LCO 3.3.4 Four ESFAS sensor modules, associated measurement channels, and applicable automatic block removal features for each Function in Table 3.3.4-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each ESFAS Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one ESFAS sensor module or associated measurement channel inoperable.	A.1 Place affected sensor module in bypass or trip.	1 hour
	AND	
	A.2.1 Restore affected sensor module and associated measurement channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
	OR	
	A.2.2 Place affected sensor module in trip.	48 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with two ESFAS sensor modules or associated measurement channels inoperable.	B.1 Place one sensor module in bypass and place the other sensor module in trip.	1 hour
	AND	
	B.2 Restore one sensor module and associated measurement channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
C. One or more Functions with the automatic block removal feature of one sensor block module inoperable.	C.1 Disable affected sensor block module.	1 hour
	OR	
	C.2 Place affected sensor block module in bypass.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more Functions with the automatic block removal feature of two sensor block modules inoperable.	D.1	Disable affected sensor block modules.	1 hour
	<u>OR</u>		
	D.2.1	Place one affected sensor block module in bypass and disable the other for each affected ESFAS Function.	1 hour
	<u>AND</u>		
	D.2.2	Restore one automatic block removal feature and the associated sensor block module to OPERABLE status for each affected ESFAS Function.	48 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2	Be in MODE 4.	12 hours

3.3 INSTRUMENTATION

3.3.5 Engineered Safety Features Actuation System (ESFAS) Logic and Manual Actuation

LCO 3.3.5 Two ESFAS Manual Actuation or Start channels and two ESFAS Actuation Logic channels shall be OPERABLE for each ESFAS Function specified in Table 3.3.5-1.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each ESFAS Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Auxiliary Feedwater Actuation System Manual Start channel or Actuation Logic channel inoperable.	A.1 Restore affected Auxiliary Feedwater Actuation System Manual Start channel and Actuation Logic channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
C ~~B~~. Required Action and associated Completion Time of Condition A or B not met.	C ~~B~~.1 Be in MODE 3. AND C ~~B~~.2 Be in MODE 4.	6 hours 12 hours
B. -- NOTE -- -- -- Not applicable when Second AFAS Manual Start channel or Actuation Logic channel intentionally made inoperable Two AFAS Manual Start channels or Actuation Logic channels inoperable.	B.1 Restore channel to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C~~ D. One or more Functions with one Manual Actuation channel or Actuation Logic channel inoperable except Auxiliary Feedwater Actuation System.	~~C~~ D.1 Restore affected Manual Actuation channel and Actuation Logic channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
~~D~~ F. Required Action and associated Completion Time of Condition ~~C~~ D or E not met for one Manual Actuation channel.	~~D~~ F.1 Be in MODE 3. AND ~~D~~ F.2 Be in MODE 5.	6 hours 36 hours
~~E~~ G. Required Action and associated Completion Time of Condition ~~C~~ D or E not met for one Actuation Logic channel.	~~E~~ G.1 Be in MODE 3. AND ~~E~~ G.2 Be in Mode 4.	6 hours 12 hours
E. -- NOTE -- Not applicable when second Manual Actuation channel or Actuation Logic channel intentionally made inoperable. One or more Functions with two Manual Actuation channels or Actuation Logic channels inoperable except AFAS.	E.1 Restore channel to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

3.3 INSTRUMENTATION

3.3.6 Diesel Generator (DG)-Loss of Voltage Start (LOVS)

LCO 3.3.6 Four sensor modules and measurement channels per DG for the Loss of Voltage Function, four sensor modules and measurement channels per DG for the Transient Degraded Voltage Function, and four sensor modules and measurement channels per DG for the Steady State Degraded Voltage Function shall be OPERABLE.

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

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one sensor module or associated measurement channel per DG inoperable.	A.1 Place sensor module in bypass or trip.	1 hour
	AND	
	A.2.1 Restore sensor module and associated measurement channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
	OR	
	A.2.2 Place the sensor module in trip.	48 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with two sensor modules or associated measurement channels per DG inoperable.	B.1 Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG-LOVS instrumentation.	1 hour
	<u>OR</u>	
	B.2.1 Place one sensor module in bypass and the other sensor module in trip.	1 hour
	<u>AND</u>	
	B.2.2 Restore one sensor module and associated measurement channel to OPERABLE status.	48 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
C. One or more Functions with more than two sensor modules or associated measurement channels inoperable.	C.1 Restore at least two sensor modules and associated measurement channels to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2).

---------------------------- NOTE ----------------------------
The lift settings are not required to be within Limiting Condition for Operation limits during MODE 3 > 365°F (Unit 1), > 301°F (Unit 2) for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 > 365°F (Unit 1), > 301°F (Unit 2) provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve. AND B.2 Remove power from associated block valve. AND B.3 Restore PORV to OPERABLE status.	1 hour 1 hour 5 days OR In accordance with the Risk Informed Completion Time Program
C. One block valve inoperable.	C.1 Place associated PORV in override closed. AND C.2 Restore block valve to OPERABLE status.	1 hour 5 days OR In accordance with the Risk Informed Completion Time Program
D. Two PORVs inoperable and not capable of being manually cycled.	D.1 Close associated block valves. AND D.2 Remove power from associated block valves. AND D.3 Restore one PORV to OPERABLE status.	1 hour 1 hour 72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two block valves inoperable.	E.1 Place associated PORVs in override closed.	1 hour
	AND	
	E.2 Restore one block valve to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3.	6 hours
	AND	
	F.2 Reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2).	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Perform a CHANNEL FUNCTIONAL TEST of each PORV.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.5.1 Safety Injection Tanks (SITs)

LCO 3.5.1 Four SITs shall be OPERABLE.

OR
In accordance with the Risk Informed Completion Time Program

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SIT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One SIT inoperable for reasons other than Condition A.	B.1 Restore SIT to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 12 hours
D. Two or more SITs inoperable.	D.1 ~~Enter LCO 3.0.3.~~	~~Immediately~~

----- NOTE -----
Not applicable when two or more SITs intentionally made inoperable.

Restore SITs to OPERABLE status.

1 hour
<u>OR</u>
In accordance with the Risk Informed Completion Time Program

3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure ≥ 1750 psia.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. ~~AND~~ ~~At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.~~	A.1 Restore train(s) to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
~~B.~~ C. Required Action and associated Completion Time not met.	~~B~~ C.1 Be in MODE 3. AND ~~B~~ C.2 Reduce pressurizer pressure to < 1750 psia.	6 hours 12 hours
B. ---NOTE--- Not applicable when second ECCS train is intentionally made inoperable. ------ Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	B.1 Restore ECCS flow equivalent to 100% of a single OPERABLE ECCS train	1 hour OR In accordance with the Risk Informed Completion Time Program

3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.5.4 Refueling Water Tank (RWT)

LCO 3.5.4 The RWT shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWT boron concentration not within limits. OR RWT borated water temperature not within limits.	A.1 Restore RWT to OPERABLE status.	8 hours OR In accordance with the Risk Informed Completion Time Program
B. RWT inoperable for reasons other than Condition A.	B.1 Restore RWT to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND		
	B.3	--------NOTE ------- Air lock doors in high radiation areas may be verified locked closed by administrative means. -------------------- Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------- NOTE -------- Only applicable to penetration flow paths with two containment isolation valves and not a closed system. --------------------- One or more penetration flow paths with one containment isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>AND</u>	
	A.2 --------NOTE ------- Isolation devices in high radiation areas may be verified by use of administrative means. -------------------- Verify the affected penetration flow path is isolated.	Once per 31 days following isolation for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. -------- NOTE -------- Only applicable to penetration flow paths with two containment isolation valves and not a closed system. --------------------- One or more penetration flow paths with two containment isolation valves inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. --------- NOTE --------- Only applicable to penetration flow paths with one or more containment isolation valves and a closed system. --------------------- One or more penetration flow paths with one or more containment isolation valves inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>AND</u>	
	C.2 --------NOTE ------- Isolation devices in high radiation areas may be verified by use of administrative means. -------------------- Verify the affected penetration flow path is isolated.	Once per 31 days following isolation
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 Be in MODE 5.	36 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

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

APPLICABILITY: MODES 1 and 2.
MODE 3, except containment spray is not required to be OPERABLE when pressurizer pressure is < 1750 psia.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. One containment cooling train inoperable.	B.1 Restore containment cooling train to OPERABLE status.	7 days
C. -------- NOTE -------- Not applicable when second containment spray train intentionally made inoperable. --------------------- Two containment spray trains inoperable.	C.1 Verify LCO 3.7.8, "CREVS," is met. AND C.2 Restore at least one containment spray train to OPERABLE status.	1 hour 24 hours

OR
In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two containment cooling trains inoperable.	D.1 Restore one containment cooling train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours
------NOTE------ Not applicable when three or more trains intentionally made inoperable. -------------- F. Any combination of three or more trains inoperable.	F.1 ~~Enter LCO 3.0.3.~~ Restore containment spray train and containment cooling train to OPERABLE status.	~~Immediately~~ 1 hour OR In accordance with the Risk Informed Completion Time Program.

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	8 hours OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
~~C~~ D. -------- NOTE -------- Separate Condition entry is allowed for each MSIV. --------------------- One or more MSIVs inoperable in MODE 2 or 3.	~~C~~ D.1 Close MSIV. AND ~~C~~ D.2 Verify MSIV is closed.	8 hours Once per 7 days
C. - - NOTE - - - - Not applicable when two or more MSIVs intentionally made inoperable - - - - - - - Two MSIVs inoperable in MODE 1	C.1 Restore MSIV to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E ~~D~~. Required Action and associated Completion Time of Condition C or D not met.	E ~~D~~.1 Be in MODE 3.	6 hours
	AND	
	E ~~D~~.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify closure time of each MSIV is within limits.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.3 Auxiliary Feedwater (AFW) System

LCO 3.7.3 Two AFW trains shall be OPERABLE.

---------------------------- NOTE ----------------------------
AFW trains required for OPERABILITY may be taken out of service under administrative control for the performance of periodic testing.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.4.b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam-driven AFW pump inoperable.	A.1 Align remaining OPERABLE steam-driven pump to automatic initiating status.	72 hours
	AND	
	A.2 Restore steam-driven pump to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One motor-driven AFW pump inoperable.	B.1	Align standby steam-driven pump to automatic initiating status.	72 hours
	<u>AND</u>		
	B.2	Restore motor-driven pump to OPERABLE status.	7 days
C. Two AFW pumps inoperable.	C.1	Align remaining OPERABLE pump to automatic initiating status.	1 hour
	<u>AND</u>		
	C.2	Verify the other unit's motor-driven AFW pump is OPERABLE.	1 hour
	<u>AND</u>		
	C.3	Verify, by administrative means, the cross-tie valve to the opposite unit is OPERABLE.	1 hour
	<u>AND</u>		
	C.4	Restore one AFW pump to OPERABLE status.	72 hours

<u>OR</u>
In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One AFW train inoperable for reasons other than Condition A, B, or C.	D.1	Restore AFW train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 4.	12 hours
F. Two AFW trains inoperable.	F.1	---------NOTE ------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. -------------------- Initiate action to restore one AFW train to OPERABLE status.	Immediately

3.7 PLANT SYSTEMS

3.7.4 Condensate Storage Tank (CST)

LCO 3.7.4 The CST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST inoperable.	A.1 Verify OPERABILITY of backup water supply. AND A.2 Restore CST to OPERABLE status.	4 hours AND Once per 12 hours thereafter 7 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

3.7 PLANT SYSTEMS

3.7.5 Component Cooling (CC) System

LCO 3.7.5 Two CC loops shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CC loop inoperable.	A.1 ---------NOTE------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops--MODE 4," for shutdown cooling made inoperable by CC. -------------------- Restore CC loop to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
~~B~~ C. Required Action and associated Completion Time ~~of Condition A~~ not met.	~~B~~ C.1 Be in MODE 3. AND ~~B~~ C.2 Be in MODE 5.	6 hours 36 hours
B. --NOTE-- Not applicable when when second CC loop intentionally made inoperable. -------- Two CC loops inoperable.	B.1 Restore at least one CC train to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

3.7 PLANT SYSTEMS

3.7.6 Service Water (SRW) System

LCO 3.7.6 Two SRW subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SRW heat exchanger inoperable.	A.1 Isolate flow to one of the associated containment cooling units. ---------NOTE-------- Enter applicable Conditions and Required Actions of LCO 3.6.6, "Containment Spray and Cooling Systems," for one containment cooling train made inoperable by the heat exchanger. ---------------------	1 hour
	AND	
	A.2 Restore heat exchanger to operable status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program.

C. --- NOTE ---
Not applicable when second SRW subsystem intentionally made inoperable.
Two SRW subsystems inoperable.

C.1 Restore at least one SRW subsystem to OPERABLE status.

1 hour
OR
In accordance with the Risk Informed Completion Time Program

SRW 3.7.6

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One SRW subsystem inoperable.	B.1 --------NOTE------- Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources--Operating," for diesel generator made inoperable by SRW. -------------------- Restore SRW subsystem to OPERABLE status.	 72 hours OR In accordance with the Risk Informed Completion Time Program
D ~~C~~. Required Action and associated Completion Time ~~of Condition A or B~~ not met.	D ~~C~~.1 Be in MODE 3. AND D ~~C~~.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 ------------------NOTE----------------- Isolation of SRW flow to individual components does not render SRW inoperable. --- Verify each SRW manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	 In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.7 Saltwater (SW) System

LCO 3.7.7 Two SW subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SW subsystem inoperable.	A.1 ------- NOTES ------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," for emergency diesel generator made inoperable by SW System. 2. Enter application Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," for shutdown cooling made inoperable by SW System. -------------------- Restore SW subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program.

B. ---NOTE--- Not applicable when second SW subsystem intentionally made inoperable. --- Two SW subsystems inoperable	B.1 Restore at least on SW subsystem to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

SW 3.7.7

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C ~~B~~. Required Action and associated Completion Time ~~of Condition A~~ not met.	C ~~B~~.1 Be in MODE 3.	6 hours
	AND	
	C ~~B~~.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	---NOTE--- Isolation of SW System flow to individual components does not render SW inoperable. --- Verify each SW System manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	Verify each SW System automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	Verify each SW System pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

B. -- NOTE --
Not applicable when second CRETS train intentionally made inoperable.

Two CRETS trains inoperable in MODE 1, 2, 3 or 4

B.1 Restore at least one CRETS train to OPERABLE status

1 hour
OR
In accordance with the Risk Informed Completion Time Program

3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Temperature System (CRETS)

LCO 3.7.9 Two CRETS trains shall be OPERABLE.

---NOTE---

Only one CRETS train is required to be OPERABLE for the movement of irradiated fuel assemblies.

APPLICABILITY: MODES 1, 2, 3, 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRETS train inoperable in MODE 1, 2, 3, or 4.	A.1 Restore CRETS train to OPERABLE status.	30 days
C ~~B~~. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C ~~B~~.1 Be in MODE 3. AND C ~~B~~.2 Be in MODE 5.	6 hours 36 hours
D ~~C~~. Two CRETS trains inoperable ~~in MODE 1, 2, 3, 4 or~~ during movement of irradiated fuel assemblies.	~~D C.1 Enter LCO 3.0.3.~~ ~~AND~~ D ~~C.2~~.1 Suspend movement of irradiated fuel assemblies.	~~Immediately~~ Immediately

3.7 PLANT SYSTEMS

3.7.15 Main Feedwater Isolation Valves (MFIVs)

LCO 3.7.15 Two MFIVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

OR

In accordance with the Risk Informed Completion Time Program

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIVs inoperable.	A.1 Restore MFIV to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the closure time of each MFIV is in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.18 Atmospheric Dump Valves (ADVs)

LCO 3.7.18 Two ADV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is being relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ADV line inoperable.	A.1	Restore ADV line to OPERABLE status.	48 hours
B. Two ADV lines inoperable.	B.1	Restore one ADV line to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4 without reliance upon steam generator for heat removal.	24 hours

OR
In accordance with the Risk Informed Completion Time Program

ACTIONS

------------------------------------NOTE------------------------------------

LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required LCO 3.8.1.a offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 or SR 3.8.1.2 for required OPERABLE offsite circuits.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND	
	A.3 Restore required offsite circuit to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4.2 Perform SR 3.8.1.3 for OPERABLE DG(s).	24 hours
	AND	
	B.5 Restore DG to OPERABLE status.	14 days
C. Required Action and associated Completion Time of Required Action B.1 not met.	C.1.1 Restore both DGs on the other unit to OPERABLE status and OC DG to available status.	72 hours
	OR	
	C.1.2 Restore DG to OPERABLE status.	

OR

In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Two required LCO 3.8.1.a offsite circuits inoperable. OR One required LCO 3.8.1.a offsite circuit that provides power to the CREVS and CRETS inoperable and the required LCO 3.8.1.c offsite circuit inoperable.	G.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition G concurrent with inoperability of redundant required feature(s)
	AND	
	G.2 Restore one required offsite circuit to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Progam
H. One required LCO 3.8.1.a offsite circuit inoperable. AND One LCO 3.8.1.b DG inoperable.	----------- NOTE ------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, when Condition H is entered with no AC power source to any train. ----------------------------	
	H.1 Restore required offsite circuit to OPERABLE status.	12 hours OR In accordance with the Risk Informed Completion Time Program
	OR	
	H.2 Restore DG to OPERABLE status.	12 hours

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
I.	Two LCO 3.8.1.b DGs inoperable. OR LCO 3.8.1.b DG that provides power to the CREVS and CRETS inoperable and LCO 3.8.1.c DG inoperable.	I.1	Restore one DG to OPERABLE status.	2 hours OR In accordance with the Risk Informed Completion Time Program
~~J.~~ K.	Required Action and associated Completion Time of Condition A, C, F, G, H, ~~or~~ I, or J not met. OR Required Action and associated Completion Time of Required Action B.2, B.3, B.4.1, B.4.2, or B.5 not met. OR Required Action and associated Completion Time of Required Action E.2, E.3, E.4.1, E.4.2, or E.5 not met.	~~J~~K.1 AND ~~J~~K.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

Add Section J from pg 3.8.1.10

- - - NOTE - - - - - -
Not applicable when three or more required LCO 3.8.1.a and LCO 3.8.1.b AC sources intentionally made inoperable.
- - - - - - - - - - - -

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~K~~ J. Three or more required LCO 3.8.1.a and LCO 3.8.1.b AC sources inoperable.	~~K~~ J.1 ~~Enter LCO 3.0.3.~~ Restore required inoperable LCO 3.8.1.a and LCO 3.8.1.b AC sources to OPERABLE status.	~~Immediately~~ 1 hour OR In accordance with the Risk Informed Completion Time Program

move to pg 3.8.1-9

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
SR 3.8.1.1 through SR 3.8.1.15 are only applicable to LCO 3.8.1.a and LCO 3.8.1.b AC sources. SR 3.8.1.16 is only applicable to LCO 3.8.1.c AC sources.

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	--------------------NOTE-------------------- Only required to be performed when SMECO is being credited for an offsite source. -- Verify correct breaker alignment and indicated power availability for the 69 kV SMECO offsite circuit.	Once within 1 hour after substitution for a 500 kV offsite circuit <u>AND</u> In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources-Operating

LCO 3.8.4 Four channels of DC electrical sources shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC channel inoperable due to an inoperable battery and the reserve battery available.	A.1 Replace inoperable battery with reserve battery.	4 hours
B. One DC channel inoperable for reasons other than Condition A.	B.1 Restore DC channel to OPERABLE status.	2 hours OR In accordance with the Risk Informed Completion Time Program
~~C.~~ D. Required Action and associated Completion Time not met.	~~C~~ D.1 Be in MODE 3. AND ~~C~~ D.2 Be in MODE 5.	6 hours 36 hours
C. -- NOTE --- Not applicable when four channels of DC electrical sources intentionally made inoperable ----------- Four channels of DC electrical sources inoperable	C.1 Restore at least three channels of DC electrical sources to OPERABLE status	1 hour OR In accordance with the Risk Informed Completion Time Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters-Operating

LCO 3.8.7 Four inverters shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 --------NOTE------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems-Operating" with any vital bus de-energized. -------------------- Restore inverter to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Program
~~B~~ C. Required Action and associated Completion Time not met.	~~B~~ C.1 Be in MODE 3. AND ~~B~~ C.2 Be in MODE 5.	6 hours 36 hours
B. ---NOTE--- Not applicable when two or more required inverters intentionally made inoperable. ------ Two or more required inverters inoperable.	B.1 Restore inverters to OPERABLE status	1 hour OR In accordance with the Risk Informed Completion Time Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems-Operating

LCO 3.8.9 The AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE.

OR
In accordance with the Risk Informed Completion Time Program

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystems inoperable.	A.1 Restore AC electrical power distribution subsystems to OPERABLE status.	8 hours
B. One or more AC vital bus subsystem(s) inoperable.	B.1 Restore AC vital bus subsystems to OPERABLE status.	2 hours
C. One DC electrical power distribution subsystem inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours
~~D~~ E. Required Action and associated Completion Time not met.	~~D~~ E.1 Be in MODE 3. AND ~~D~~ E.2 Be in MODE 5.	6 hours 36 hours

move Section E to page 3.8.9-2

- NOTE -
Not applicable when two or more electrical power distribution subsystems intentionally made inoperable.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E~~D. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	~~E~~D.1 ~~Enter LCO 3.0.3.~~ Restore electrical power distribution subsystems to OPERABLE status.	~~Immediately~~ 1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

Section E from pg 3.8.9-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to AC, DC, and AC vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals

the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and assessing the CRE boundary as required by paragraphs c and d respectively.

5.5.18 ~~Not Used~~ Insert 2

5.5.19 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

Insert 2

5.5.18 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09, Revision 0-A, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1, and 2;

c. When a RICT is being used, any plant configuration change within the scope of the Risk Informed Completion Time Program must be considered for the effect on the RICT.

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. Use of a RICT is not permitted for voluntary entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

e. Use of a RICT is permitted for emergent conditions which represent a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE if one or more of the trains are considered "PRA functional" as defined in Section 2.3.1 of NEI 06-09.

ATTACHMENT 3

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Proposed Technical Specification Bases Changes (Mark-Ups)

Insert 1

or in accordance with the Risk Informed Completion Time Program

Insert 2

Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

Insert 3 (for TS 3.3.5)

<u>B.1</u>
If two AFAS Manual Trip or Actuation Logic channels are inoperable, the Required Action is to restore at least one channel to OPERABLE status within 1 hour. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one channel. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when the second AFAS Manual Trip or Actuation Logic channel is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one AFAS Manual Trip or Actuation Logic channel is inoperable for any reason and a second AFAS Manual Trip or Actuation Logic channel is found to be inoperable, or if two AFAS Manual Trip or Actuation Logic channels are found to be inoperable at the same time.

Insert 4 (for TS 3.3.5)

<u>E.1</u>
If one or more Functions have two Manual Trip or Actuation Logic channels inoperable except AFAS, the Required Action is to restore the Functions to OPERABLE status within 1 hour. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of the Manual Trip or Actuation Function Logic Function. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when the second Manual Trip or Actuation Logic channel is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one Manual Trip or Actuation Logic channel is inoperable for any reason and a second Manual Trip or Actuation Logic channel is found to be inoperable, or if two Manual Trip or Actuation Logic channels are found to be inoperable at the same time.

Insert 5 (for TS 3.5.1)

<u>D.1</u>
With two or more SITs inoperable, the Required Action is to restore sufficient SITs to OPERABLE status within 1 hour to regain this safety function. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of sufficient SITs to regain safety function. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when two or more SITs are intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one SIT is inoperable for any reason and additional SITs are found to be inoperable, or if two or more SITs are found to be inoperable at the same time.

Insert 6 (for TS 3.5.2)

B.1
Condition A is for one or more trains inoperable. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the facility is in a condition outside the accident analyses and flow must be restored to 100% of the ECCS flow equivalent to a single OPERABLE ECCS train within the 1 hour Completion Time, or a Completion Time determined under the Risk Informed Completion Time Program. The Completion Time is based on the need to restore the ECCS flow to within the safety analyses assumptions.

The Condition is modified by a Note stating it is not applicable when the second ECCS train is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one ECCS train is inoperable for any reason and the second ECCS train is found to be inoperable, or if two ECCS trains are found to be inoperable at the same time.

Insert 7 (for TS 3.6.6)

F.1
With any combination of three or more trains inoperable, sufficient containment spray trains and/or containment cooling trains must be restored to OPERABLE status so that no more than one containment spray train or two containment cooling trains are inoperable within one hour or in accordance with the Risk Informed Completion Time Program. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of sufficient trains.

The Condition is modified by a Note stating it is not applicable when three or more trains are intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one containment spray train or any combination of two containment spray and cooling trains are inoperable for any reason and a second containment spray train or additional containment spray or cooling trains are found to be inoperable, or if two containment spray trains or any combination of three or more containment spray and cooling trains are found to be inoperable at the same time.

Insert 8 (for TS 3.7.2)

C.1
With two MSIVs inoperable, the Required Action is to restore sufficient required MSIVs to OPERABLE status within 1 hour to regain a method of main steam line isolation. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of sufficient required MSIVs. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when two MSIVs are intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one MSIV is inoperable for any reason and an additional MSIV is found to be inoperable, or if two MSIVs are found to be inoperable at the same time.

Insert 9 (TS 3.7.5)

B.1
With two CC loops inoperable, the Required Action is to restore at least one of the required CC loops to OPERABLE status within 1 hour to regain a heat sink for safety related components. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one loop. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when the second CC loop is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one CC loop is inoperable for any reason and a second CC loop is found to be inoperable, or if two CC loops are found to be inoperable at the same time.

Insert 10 (for TS 3.7.6)

C.1
With two SRW subsystems inoperable, the Required Action is to restore at least one of the required SRW subsystems to OPERABLE status within 1 hour to regain a heat sink for safety related components. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one loop. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when the second SRW subsystem is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one SRW subsystem is inoperable for any reason and a second SRW subsystem is found to be inoperable, or if two SRW subsystems are found to be inoperable at the same time.

Insert 11 (for TS 3.7.7)

B.1
With two SW subsystems inoperable, the Required Action is to restore at least one of the required SW subsystems to OPERABLE status within 1 hour to regain a heat sink for safety related components. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one loop. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when the second SW subsystem is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one SW subsystem is inoperable for any reason and a second SW subsystem is found to be inoperable, or if two SW subsystems are found to be inoperable at the same time.

Insert 12 (for TS 3.7.9)

B.1
With two CRETS trains inoperable, the Required Action is to restore at least one of the required CRETS trains to OPERABLE status within 1 hour to regain temperature control for the control room following isolation of the control room. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one loop. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when the second CRETS train is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one CRETS train is inoperable for any reason and a second CRETS train is found to be inoperable, or if two CRETS trains are found to be inoperable at the same time.

Insert 13 (for TS 3.8.1)

J.1
With three or more required AC sources inoperable, the Required Action is to restore enough of the required inoperable AC sources to OPERABLE status within 1 hour to regain some level of redundancy in the AC electrical power supplies. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of sufficient AC sources. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when three or more required AC sources are intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if two required AC sources are inoperable for any reason and additional required AC sources are found to be inoperable, or if three or more required AC sources are found to be inoperable at the same time.

Insert 14 (for TS 3.8.4)

C.1
With four DC electrical source channels inoperable, the Required Action is to restore at least three of the required DC electrical source channels to OPERABLE status within 1 hour to regain control power for the AC emergency power system. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least three required DC electrical source channels. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when four DC electrical source channels are intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one DC electrical source channel is inoperable for any reason and three DC electrical source channels are found to be inoperable, or if four DC electrical source channels are found to be inoperable at the same time.

Insert 15 (for TS 3.8.7)

B.1

With two required inverters inoperable, the Required Action is to restore at least one of the required inverters to OPERABLE status within 1 hour to regain AC electrical power to the vital busses. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one required inverter. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when two or more required inverters are intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one required inverter is inoperable for any reason and additional required inverters are found to be inoperable, or if two or more required inverters are found to be inoperable at the same time.

Insert 16 (for TS 3.8.9)

D.1

With two or more electrical power distribution subsystems inoperable that result in a loss of safety function, the Required Action is to restore sufficient electrical power distribution subsystems within 1 hour to restore safety function. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of safety function. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by a Note stating it is not applicable when two or more electrical power distribution subsystems are intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one electrical power distribution subsystem is inoperable for any reason and a second electrical power distribution subsystem is found to be inoperable, or if two or more electrical power distribution subsystems are found to be inoperable at the same time.

BASES

to continue, providing the inoperable bistable trip unit is placed in bypass or trip within 1 hour (Required Action A.1).

The Completion Time of 1 hour allotted to restore, bypass, or trip the instrument channel is sufficient to allow the operator to take all appropriate actions for the failed channel, while ensuring that the risk involved in operating with the failed channel is acceptable.

The failed instrument channel is restored to OPERABLE status or is placed in trip within 48 hours Insert 1 (Required Action A.2.1 or Required Action A.2.2). Required Action A.2.1 restores the full capability of the Function.

Required Action A.2.2 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent a trip.

The Completion Time of 48 hours is based on operating experience, which has demonstrated that a random failure of a second instrument channel occurring during the 48-hour period is a low probability event.

B.1 and B.2

Condition B applies to the failure of two instrument channels in any RPS automatic trip Function.

Required Action B.1 provides for placing one inoperable channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels, while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed, the RPS Function is in a two-out-of-three logic; but with another channel failed, the RPS Function may be operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS Function in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

One instrument (Insert 1) channel should be restored to OPERABLE status within 48 hours for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than 48 hours have elapsed since the initial channel failure.

C.1 and C.2

The excore detectors are used to generate the internal ASI used as an input to the TM/LP and APD-High trips. Incore detectors provide a more accurate measurement of ASI. If one or more excore channels cannot be calibrated to match incore detectors, power is restricted or reduced during subsequent operations because of increased uncertainty associated with using uncalibrated excore channels.

The Completion Time of 24 hours is adequate to perform the Surveillance Requirement (SR) while minimizing the risk of operating in an unsafe condition.

D.1, D.2.1, D.2.2.1, and D.2.2.2

Condition D applies to one automatic bypass removal feature inoperable. If the automatic bypass removal feature for any operating bypass channel cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the bypass either removed or the automatic bypass removal feature repaired. The Bases for Required Actions and Completion Times are the same as discussed for Condition A.

E.1, E.2.1, and E.2.2

Condition E applies to two inoperable automatic bypass removal features. If the automatic bypass removal features cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypasses are

BASES

any RTCB is closed and any CEA is capable of being withdrawn.

Two independent sets of two adjacent push buttons are provided at separate locations. Each push button is considered a channel and operates two of the eight RTCBs. Depressing both push buttons in either set will cause an interruption of power to the CEDMs, allowing the CEAs to fall into the core. This design ensures that no single failure in any push button channel can either cause or prevent a reactor trip.

APPLICABILITY

The RPS matrix logic, RTCBs, and manual trip are required to be OPERABLE in any MODE when any CEA is capable of being withdrawn from the core (i.e., RTCBs closed and power available to the CEDMs). This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

In MODEs 3, 4, and 5, with all the RTCBs open, the CEAs are not capable of withdrawal and these Functions do not have to be OPERABLE. However, two wide range logarithmic neutron flux monitor channels must be OPERABLE to ensure proper indication of neutron population and to indicate a boron dilution event. This is addressed in LCO 3.3.12.

ACTIONS

When the number of inoperable RPS logic or trip initiation channels exceeds that specified in any related Condition, the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies if one matrix logic channel is inoperable or three logic matrices channels are inoperable due to a common power source failure de-energizing three matrix power supplies in any applicable MODE.

Insert 1

The matrix logic channel must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed

3. <u>CIS</u>

   Containment Pressure-High Trip

4. <u>SGIS</u>

   Steam Generator Pressure-Low Trip

5. <u>RAS for the Containment Sump</u>

   RWT Level-Low Trip

6. <u>AFAS Signal</u>

   Steam Generator Level-Low Trip
   Steam Generator Pressure Difference-High Trip

Engineered Safety Features Actuation System coincidence logic is normally two-out-of-four. If one ESFAS sensor channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESFAS sensor channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action A.1).

The Completion Time of 1 hour allotted to bypass or trip the sensor channel is sufficient to allow the operator to take all appropriate actions for the failed channel, and still ensures that the risk involved in operating with the failed channel is acceptable.

Insert 1

One failed sensor channel is restored to OPERABLE status or is placed in trip within 48 hours (Required Action A.2.1 or A.2.2). Required Action A.2.1 restores the full capability of the function. Required Action A.2.2 places the function in a one-out-of-three configuration. In this configuration, common cause failure of the dependent channel cannot prevent ESFAS actuation. The 48-hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48-hour period is a low probability event.

Insert 2

B.1 and B.2

Condition B applies to the failure of two sensor channels in any of the following ESFAS functions:

1. SIAS

   Containment Pressure-High Trip
   Pressurizer Pressure-Low Trip

2. CSAS

   Containment Pressure-High Trip

3. CIS

   Containment Pressure-High Trip

4. SGIS

   Steam Generator Pressure-Low Trip

5. RAS for the Containment Sump

   RWT Level-Low Trip

6. AFAS Signal

   Steam Generator Level-Low Trip
   Steam Generator Pressure Difference-High Trip

With two inoperable sensor channels, one channel should be placed in bypass, and the other channel should be placed in trip within the 1-hour Completion Time. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic; but with another channel failed, the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS in a one-out-of-two logic. If any of the other OPERABLE channels receive a trip signal, ESFAS actuation will occur.

Insert 1

One of the failed sensor channels should be restored to OPERABLE status within 48 hours. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than 48 hours has elapsed since the initial channel failure.

Insert 2

C.1 and C.2

Condition C applies to the failure of one automatic block removal feature when the block is in effect.

The automatic block removal features are incorporated into the four sensor block modules (per steam generator for SGIS) and two block logic modules. Condition C applies to failures in the automatic block removal feature of one of the four sensor block modules. Failures in the block logic modules, including the block logic manual bypass key switches, are considered actuation logic failures and are addressed in LCO 3.3.5.

In Condition C, it is permissible to continue operation with the automatic block removal feature in one sensor block module failed, providing the sensor block module is disabled (Required Action C.1). This can be accomplished by adjusting the sensor block module setpoint, which disables the sensor block modules to both block logic modules. Therefore, a block permissive signal is not produced by the sensor block module.

Placing a sensor module in bypass defeats the block permissive input in one of the four channels to the two-out-of-four block removal logic, placing the automatic block removal feature in one-out-of-three logic. Thus, any of the remaining three channels is capable of removing the block feature when the block enable conditions are no longer valid.

In this configuration, common cause failure of the dependent channel cannot prevent block removal.

D.1, D.2.1, and D.2.2

Condition D applies to two inoperable automatic block removal features. The automatic block removal features consist of four sensor block modules (per steam generator for SGIS) and two actuation logic channels. This Condition applies to failures in two of the four sensor block modules. With two of the four sensor block modules failed in a nonconservative direction (enabling the block feature), the automatic block removal feature is in two-out-of-two logic. Failures in the actuation logic channels, including the manual bypass key switches, are considered actuation logic failures and are addressed in LCO 3.3.5.

In Condition D, it is permissible to continue operation with two automatic block removal features failed, providing the sensor block modules are disabled in a similar manner as discussed for Condition C.

If the failed sensor block modules cannot be disabled, actions to address the inoperability of the affected sensor block modules must be taken. Required Action D.2.1 and Required Action D.2.2 are equivalent to the Required Actions for a two sensor channel failure (Condition B). Also similar to Condition B, after one inoperable sensor block module is restored, the provisions of Condition C still apply to the remaining inoperable automatic block removal feature, with the Completion Time measured from the point of the initial bypass channel failure. The 1-hour Completion Time minimizes the time that the plant is in two-out-of-two logic. The 48-hour Completion Time limits the time the plant is in one-out-of-two logic. Limits on the time in these logic conditions are similar to those found in Action B. Insert 2

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, C, or D are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to 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.

BASES

The ESFAS actuation logic must be OPERABLE in the same MODEs as the automatic and manual actuations. In MODE 4, only the portion of the ESFAS logic responsible for the required manual actuation must be OPERABLE.

In MODEs 5 and 6, ESFAS actuated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODEs are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

When the number of inoperable actuation logic or manual actuation channels in an ESFAS Function exceeds those specified in any related Condition associated with the same ESFAS Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.5-1 in the LCO. Completion Times for the inoperable actuation logic channel of a Function will be tracked separately.

A.1

Condition A applies to one AFAS manual actuation or AFAS actuation logic channel inoperable. It is identical to Condition C for the other ESFAS Functions, except for the shutdown track imposed by Condition D.

The channel must be restored to OPERABLE status to restore redundancy of the AFAS Function. The 48-hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel. Insert 2

Insert 3

C ~~B~~.1 and C ~~B~~.2

If the Required Action and associated Completion Time of Condition A or B cannot be met, the reactor should be brought to a mode in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed

BASES

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.~~C~~.1

Condition ~~C~~ D applies to one manual actuation or actuation logic channel inoperable for those ESFAS Functions that must be OPERABLE in MODEs 1, 2, 3, and 4 (manual actuation) or MODEs 1, 2, and 3 (actuation logic channel). Actuation logic includes the block logic modules when the affected block is in effect. The shutdown track imposed by Condition ~~D~~ or ~~E~~ requires entry into MODE 4 or 5, respectively, where the LCO does not apply to the affected Functions. Insert 2

The channel must be restored to OPERABLE status to restore redundancy of the affected Functions. The 48-hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel.

Insert 4

F ~~D~~.1 and F ~~D~~.2

Condition ~~D~~ F is entered when the Required Action and associated Completion Time of Condition ~~C~~ D are not met for one manual actuation channel. If Required Action D ~~C~~.1 for one manual actuation channel cannot be met within the required Completion Time, the plant must be brought to a mode in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours to MODE 5 within 36 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.

G ~~E~~.1 and G ~~E~~.2

Condition ~~E~~ G is entered when the Required Action and associated Completion Time of Condition ~~C~~ D are not met for one actuation logic channel. If Required Action D ~~C~~.1 for one actuation logic channel cannot be met within the required Completion Time, the plant must be brought to a MODE in

In the event a sensor channel's setting is found to be nonconservative with respect to the Allowable Value, or the channel is found to be inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered. The required channels are specified on a per DG basis.

When the number of inoperable channels in a Function exceeds those specified in any related Condition associated with the same Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be entered immediately if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this LCO may be entered independently for each Function. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1, A.2.1, and A.2.2

Condition A applies if one sensor channel is inoperable for one or more Functions per DG bus.

If the channel cannot be restored to OPERABLE status, the affected channel should either be bypassed or tripped within 1 hour (Required Action A.1).

Placing this channel in either Condition ensures that logic is in a known configuration. In trip, the LOVS logic is one-out-of-three. In bypass, the LOVS logic is two-out-of-three. The 1-hour Completion Time is sufficient to perform these Required Actions.

Insert 2

Once Required Action A.1 has been complied with, Required Action A.2.1 allows 48 hours to repair the inoperable sensor channel. If the channel cannot be restored to OPERABLE status, it must be tripped in accordance with Required Action A.2.2. The time allowed to repair or trip the channel is reasonable to repair the affected channel while ensuring that the risk involved in operating with the inoperable channel is acceptable. The 48-hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel is a rare event during any given 48-hour period. Insert 2

B.1, B.2.1, and B.2.2

Condition B applies if two sensor channels are inoperable for one or more Functions per DG.

Restoring at least one channel to OPERABLE status is the preferred action. If the channel cannot be restored to OPERABLE status within 1 hour, the Conditions and Required Actions for the associated DG made inoperable by DG-LOVS instrumentation are required to be entered. Alternatively, one affected channel is required to be bypassed and the other is tripped, in accordance with Required Action B.2.1. This places the Function in one-out-of-two logic. The 1-hour Completion Time is sufficient to perform the Required Actions.

Once Required Action B.2.1 has been complied with, Required Action B.2.2 allows 48 hours to repair the bypassed or inoperable channel. Insert 2

After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2.2 shall be placed in trip if more than 48 hours have elapsed since the initial channel failure. Insert 2

C.1

Condition C applies when more than two undervoltage or degraded (transient or steady state) voltage sensor channels on a single bus are inoperable.

Insert 1

Required Action C.1 requires all but two channels to be restored to OPERABLE status within 1 hour. With more than two channels inoperable, the logic is not capable of providing a DG-LOVS signal for valid loss of voltage or degraded voltage conditions. The 1 hour Completion Time is reasonable to evaluate and take action to correct the

The Note allows entry into MODE 3 > 365°F (Unit 1), > 301°F (Unit 2) with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high P/T near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed within this time frame.

ACTIONS

A.1

Insert 1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if two pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and at or below 365°F (Unit 1), 301°F (Unit 2) with all RCS cold leg temperatures ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) within 12 hours. The six hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power without challenging plant systems. Similarly, the 12 hours allowed is reasonable, based on operating experience, to reduce temperature to below 365°F (Unit 1), 301°F (Unit 2) without challenging plant systems. At or below 365°F (Unit 1), 301°F (Unit 2), overpressure protection is provided by LTOP. The change from MODEs 1 or 2, or MODE 3 > 365°F (Unit 1), > 301°F (Unit 2) to MODE 3 ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

A.1

With one or two PORVs inoperable and capable of being manually cycled, either the inoperable PORV(s) must be restored or the flow path isolated within one hour. The block valve should be closed but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. Although the PORV may be designated inoperable, it may be able to be manually opened and closed, and in this manner can be used to perform its function. Power-operated relief valve inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use, and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of one hour is based on plant operating experience that minor problems can be corrected or closure can be accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable and not capable of being manually cycled, it must either be isolated, by closing the associated block valve and removing the power from the block valve, or restored to OPERABLE status. The Completion Time of one hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, five days are provided to restore the inoperable PORV to OPERABLE status. Insert 2

C.1 and C.2

If one block valve is inoperable, then it must be restored to OPERABLE status, or the associated PORV placed in override closed. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within one hour, the Required Action is to place the PORV in override closed to preclude its automatic opening for an overpressure event, and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Times of one hour are reasonable based on the small potential for challenges to the system during this time period and provide the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of five days to restore the inoperable block valve to OPERABLE status. Insert 2 The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B since the PORVs are not capable of automatically mitigating an overpressure event when placed in override closed. If the block valve is restored within the Completion Time of five days, the power will be restored and the PORV restored to OPERABLE status.

D.1, D.2, and D.3

If both PORVs are inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of one hour Insert 1 or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of one hour is reasonable based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If Required Actions D.1 and D.2 have been completed, Required Action D.3 allows 72 hours to restore a PORV to OPERABLE status. This time is reasonable to perform required repairs. This time also accounts for the overpressure protection provided by the pressurizer safety valves in LCO 3.4.10.

E.1 and E.2

If both block valves are inoperable, it is necessary to either restore the block valves within the Completion Time

BASES

Insert 1

of one hour or place the associated PORVs in override closed and restore at least one block valve to OPERABLE status within 72 hours, and the remaining block valve in five days, per Required Action C.2. The Completion Time of one hour to either restore the block valves or place the associated PORVs in override closed is reasonable based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) within 12 hours. The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) is reasonable considering that a plant can cool down within that time frame. In MODE 3 with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) and in MODEs 4, 5, and 6, maintaining PORV OPERABILITY is required per LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

A CHANNEL FUNCTIONAL TEST is performed on each PORV instrument channel to ensure the entire channel will perform its intended function when needed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.11.2

Block valve cycling verifies that it can be closed if necessary. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance because opening the block valve is necessary to permit the PORV to be used for manual control of RCS allows RCS cooldown and depressurization without discharging the SITs into the RCS or requiring depressurization of the SITs.

ACTIONS

A.1

Insert 1

If the boron concentration of one SIT is not within limits it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced, but the reduced concentration effects on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the SIT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of three SITs, the consequences are less severe than they would be if an SIT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one SIT is inoperable, for reasons other than boron concentration, the SIT must be returned to OPERABLE status within one hour. In this Condition, the required contents of three SITs cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the one hour Completion Time to open the valve, remove power from the valve, or restore proper water volume or nitrogen cover pressure, ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the exposure of the plant to a LOCA in these conditions.

Insert 1

C.1 and C.2

If the SIT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power

BASES

conditions in an orderly manner and without challenging plant systems.

replace with Insert 5

~~D.1~~

~~If more than one SIT is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.~~

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification that each SIT isolation valve is fully open, as indicated in the Control Room, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor-operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.2 and SR 3.5.1.3

Safety injection tank borated water volume and nitrogen cover pressure should be verified to be within specified limits in order to ensure adequate injection during a LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.4

Six months is reasonable for verification by sampling to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Verification consists of monitoring inleakage or sampling. The inleakage is monitored by monitoring tank level. Sampling of each tank is done. All intentional sources of level increase are maintained administratively to ensure SIT boron concentrations are within technical specification limits. The boron concentration of each tank is verified

BASES

The HPSI pump performance is based on the small break LOCA, which establishes the pump performance curve and has less dependence on power. The requirements of MODE 2, and MODE 3 with RCS pressure ≥ 1750 psia, are bounded by the MODE 1 analysis.

The ECCS functional requirements of MODE 3, with RCS pressure < 1750 psia, and MODE 4 are described in LCO 3.5.3.

In MODEs 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7 and LCO 3.4.8. MODE 6 core cooling requirements are addressed by LCO 3.9.4 and LCO 3.9.5.

ACTIONS

A.1

If one or more trains are inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, the inoperable components must be returned to OPERABLE status within 72 hours. Insert 1 The 72 hour Completion Time is based on an Nuclear Regulatory Commission study (Reference 3) using a reliability evaluation and is a reasonable amount of time to effect many repairs.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS. The individual components are inoperable if they are not capable of performing their design function, or if supporting systems are not available.

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

An event accompanied by a loss of offsite power and the failure of an emergency diesel generator can disable one ECCS train until power is restored. A reliability analysis (Reference 3) has shown that the impact with one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 4 describes situations in which one component, such as a SDC total flow control valve, can disable both ECCS trains. With one or more components inoperable, such that 100% of the equivalent flow to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, ~~LCO 3.0.3~~ Condition B must be immediately entered.

Insert 6

~~B~~C.1 and ~~B~~C.2

If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 1750 psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. MOV-659 and MOV-660 are secured in position by interrupting the control signal to the valve operator via a key switch in the Control Room. Power is removed from the valve operator for CV-306 by isolating the air supply to the valve positioner. These actions ensure that the valves cannot be inadvertently misaligned. These valves are of the type described in Reference 4, which can disable the function of both ECCS trains and invalidate the accident analysis. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

LOCA, ensure that the reactor remains subcritical following a DBA, and ensure that an adequate level exists in the containment sump to support ESF pump operation in the recirculation mode.

To be considered OPERABLE, the RWT must meet the limits established in the SRs for water volume, boron concentration, and temperature.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the RWT OPERABILITY requirements are dictated by the ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODEs 1, 2, 3, and 4, the RWT must be OPERABLE to support their operation.

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7 and LCO 3.4.8. MODE 6 core cooling requirements are addressed by LCO 3.9.4 and LCO 3.9.5.

ACTIONS

A.1

With RWT boron concentration or borated water temperature not within limits, it must be returned to within limits within eight hours. Insert 1 In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of eight hours to restore the RWT to within limits was developed considering the time required to change boron concentration or temperature, and that the contents of the tank are still available for injection.

Required Action A.1 only applies to the maximum borated water temperature in MODE 1.

B.1

With RWT borated water volume not within limits, it must be returned to within limits within one hour. Insert 1 In this condition, neither the ECCS nor Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the unit in a MODE in which these systems are not required. The allowed Completion Time of one hour to administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Conditions A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the Containment Structure inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), the Containment Structure remains OPERABLE, yet only one hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the one hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that the Containment Structure be restored to OPERABLE status within one hour.

Insert D

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

The fourth Note has been added that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1, when leakage results in exceeding the overall containment leakage limit.

The fifth Note allows the shutdown cooling isolation valves to be opened when RCS temperature is < 300°F to establish shutdown cooling flow. This Note is required for Operation in MODE 4 to allow shutdown cooling to be established.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to the Containment Structure. Required Action A.1 must be completed within the four hour Completion Time. Insert 1 The four hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting the Containment Structure OPERABILITY during MODEs 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the four hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it following isolation

involves verification, through a system walkdown, that those isolation devices outside the Containment Structure and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside Containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside the Containment Structure, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves and not a closed system. For penetration flow paths with one or more containment isolation valves and a closed system, Condition C provides appropriate actions.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1

Insert 1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within one hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The one hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required

Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of the Containment Structure and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated, is appropriate, considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more containment isolation valves inoperable in one or more penetration flow paths, the inoperable valves must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 72 hour Completion Time. Insert 2 The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting the Containment Structure OPERABILITY during MODEs 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to assure leak tightness of the Containment Structure and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days following isolation for verifying that each affected penetration flow path is isolated, is appropriate

The Containment Spray System is only required to be OPERABLE in MODE 3 with pressurizer pressure ≥ 1750 psia.

In MODE 3 with pressurizer pressure < 1750 psia, and in MODEs 4, 5, and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODEs. Thus, the Containment Spray System is not required to be OPERABLE in MODE 3 with pressurizer pressure < 1750 psia, and the Containment Spray and Cooling Systems are not required to be OPERABLE in MODEs 4, 5, and 6.

ACTIONS

A.1

Insert 1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.1

Insert 1

With one required containment cooling train inoperable, the inoperable containment cooling train must be restored to OPERABLE status within seven days. The remaining OPERABLE containment spray and cooling components are capable of providing greater than 100% of the heat removal needs (for the condition of one containment cooling train inoperable) after an accident. The seven day Completion Time was developed based on the same reasons as those for Required Action A.1.

C.1 and C.2

Insert 1

With two required containment spray trains inoperable, at least one of the required containment spray trains must be restored to OPERABLE status within 24 hours. Both trains of containment cooling must be OPERABLE or Condition F is also entered. The Condition is modified by a Note stating it is not applicable if the second containment spray train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. In addition, LCO 3.7.11, CREVS, must be verified to be met within one hour. The OPERABLE containment cooling system components are capable of providing greater than 100% of the heat removal needs after an accident. The Completion Time is based on Reference 2 which demonstrated that the 24 hour Completion Time is acceptable based on the redundant heat removal capabilities afforded by the Containment Cooling System, the iodine removal capability of the Control Room Emergency Ventilation System, the infrequent use of the Required Action, and the small incremental effect on plant risk.

D.1

Insert 1

With two required containment cooling trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. The remaining OPERABLE containment spray components provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray and Cooling Systems, the iodine removal function of the Containment Spray System, and the low probability of a DBA occurring during this period.

E.1 and E.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to 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.

BASES

Insert 7

~~F.1~~

~~With any combination of three or more Containment Spray and Cooling Systems trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.~~

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verifying, through a system walkdown, that those valves outside the Containment Structure and capable of potentially being mispositioned are in the correct position.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.2

Starting each containment cooling train fan unit from the Control Room and operating it for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected and corrective action taken. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.3

Verifying a service water flow rate of ≥ 2000 gpm to each cooling unit when the full flow service water outlet valves are fully open provides assurance that the design flow rate assumed in the safety analyses will be achieved (Reference 1, Chapter 7). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

generator from the intact steam generator and minimizes radiological releases. The operator is then required to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

e. The MSIVs are also utilized during other events such as a feedwater line break. These events are less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

This LCO requires that the MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents as described in Reference 1, Chapter 14.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODEs 2 and 3, except when all MSIVs are closed. In these MODEs there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing their safety function.

In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.

In MODEs 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODEs.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the unit hot. The eight hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs. Insert 2

B.1

If the MSIV cannot be restored to OPERABLE status within eight hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within six hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2, and close the MSIVs in an orderly manner and without challenging unit systems.

Insert 8

~~C~~ D.1 and ~~C~~ D.2

Condition ~~C~~ D is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODEs 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The eight hour Completion Time is consistent with that allowed in Condition A.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The seven day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the Control Room, and other administrative controls, to ensure these valves are in the closed position.

E ~~D~~.1 and ~~D~~ E.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from

Insert 1

standby or automatic initiating status, depending upon whether the other steam-driven AFW pump is in standby or automatic initiating status) within seven days. The 72 hour and seven day Completion Times are reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and flow paths remain to supply feedwater to the steam generators.

B.1 and B.2

With the motor-driven AFW pump inoperable, action must be taken to align the standby steam-driven pump to automatic initiating status. This Required Action ensures that another AFW pump is available to automatically start, if required. If the standby steam-driven pump is properly aligned, the inoperable motor-driven AFW pump must be restored to OPERABLE status within seven days. Insert 1 The 72-hour and seven day, Completion Times are reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and one flow path remain to supply feedwater to the steam generators.

C.1, C.2, C.3, and C.4

With two AFW pumps inoperable, action must be taken to align the remaining OPERABLE pump to automatic initiating status and to verify the other units motor-driven AFW pump is OPERABLE, along with an OPERABLE cross-tie valve, within one hour. If these Required Actions are completed within the Completion Time, one AFW pump must be restored to OPERABLE status within 72 hours. Insert 1 Verifying the other unit's motor-driven AFW pump is OPERABLE provides an additional level of assurance that AFW will be available if needed, because the other unit's AFW can be cross-connected if necessary. The cross-tie valve to the opposite unit is administratively verified OPERABLE by confirming that SR 3.7.3.2 has been performed within the specified Frequency. These one hour Completion Times are reasonable based on the low probability of a DBA occurring during the first hour and the need for AFW during the first hour. The 72 hour completion time to restore one AFW pump to OPERABLE status takes into account the cross-connected capability between units and the unlikelihood of an event occurring in the 72 hour period.

D.1

With one of the required AFW trains inoperable for reasons other than Condition A, B, or C (e.g., flowpath or steam supply valve), action must be taken to restore OPERABLE status within 72 hours. Insert 1 This Condition includes the loss of two steam supply lines to the turbine-driven AFW pumps. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. One AFW train remains to supply feedwater to the steam generators.

E.1 and E.2

When the Required Action and associated Completion Time of Condition A, B, C, or D cannot be met the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

F.1

Required Action F.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status.

With two AFW trains inoperable in MODEs 1, 2, and 3, the unit may be in a seriously degraded condition with only non-safety-related means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. However, a power change is not precluded if it is determined to be the most prudent action. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. While other plant conditions may require entry into LCO 3.0.3, the ACTIONS usable volume and water not usable because of the tank discharge line location.

OPERABILITY of the CST is determined by maintaining the tank volume at or above the minimum required volume.

APPLICABILITY

In MODEs 1, 2, and 3, the CST is required to be OPERABLE.

In MODEs 4, 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup water supply (CST No. 11 for Unit 1 and CST No. 21 for Unit 2) must be verified by administrative means within 4 hours and once every 12 hours thereafter.

Insert 2

OPERABILITY of the backup feedwater supply must include verification that the manual valves in the flow paths from the backup supply to the AFW pumps are open, and availability of the required volume of water (150,000 gallons) in the backup supply. The CST must be returned to OPERABLE status within seven days, as the backup supply may be performing this function in addition to its normal functions. The four hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The seven day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event requiring the use of the water from the CST occurring during this period.

If the CST volume is less than 300,000 gallons and greater than 150,000 gallons and both units are in the MODE of Applicability, only one unit must enter this condition provided the unit aligns to the OPERABLE backup water supply (CST No. 11 or CST No. 21).

b. The associated piping, valves, heat exchanger and instrumentation and controls required to perform the safety-related function are OPERABLE.

The isolation of CC from other components or systems not required for safety may render those components or systems inoperable, but does not affect the OPERABILITY of the CC System.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the CC System is a normally operating system that must be prepared to perform its post accident safety functions, primarily RCS heat removal by cooling the SDC heat exchanger.

In MODEs 5 and 6, the OPERABILITY requirements of the CC System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating the requirement of entry into the applicable Conditions and Required Actions of LCO 3.4.6, for SDC made inoperable by CC. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

Insert 1

With one CC loop inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CC loop is adequate to perform the heat removal function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this period.

Insert 9

C ~~B~~.1 and C ~~B~~.2

If the CC loop cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions

BASES

ACTIONS

A.1 and A.2

Insert 1

With one SRW heat exchanger inoperable, action must be taken to restore operable status within 7 days. Isolating flow to one associated containment cooling unit will reduce the DBA heat load of the affected SRW subsystem to within the capacity of one SRW heat exchanger, thus ensuring that the SRW temperatures can be maintained within their design limits. This will allow the associated diesel generator (except for 11 SRW which does not cool a diesel generator) to remain operable. In this Condition, the other OPERABLE SRW System is adequate to perform the containment heat removal function. However, the overall reliability is reduced because a single failure in the SRW System could result in loss of SRW containment heat removal function. Required Action A.1 is modified by a Note. The Note indicates that the applicable Conditions of LCO 3.6.6 should be entered for an inoperable containment cooling train. The 7 day Completion Time is based on the redundant capabilities afforded by the OPERABLE subsystem, the Completion Time associated with an inoperable containment cooling unit (3.6.6), and the low probability of a DBA occurring during this time period.

B.1

Insert 1

With one SRW subsystem inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SRW System is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the SRW System could result in loss of SRW function. Required Action B.1 is modified by a Note. The Note indicates that the applicable Conditions of LCO 3.8.1, should be entered if the inoperable SRW subsystem results in an inoperable diesel generator. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE subsystem, and the low probability of a DBA occurring during this time period.

Insert 10

D ~~C~~.1 and D ~~C~~.2

If the SRW subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To

BASES

APPLICABLE SAFETY ANALYSES

The most limiting event for the SW System is a LOCA. Operation of the SW System following a LOCA is separated into two phases, before the RAS and after the RAS. One subsystem can satisfy cooling requirements of both phases. After a LOCA but before an RAS, each subsystem will cool two SRW heat exchangers and an ECCS pump room air cooler (as required). There is no required flow to the CC heat exchangers. When an RAS occurs, flow is throttled to the CC heat exchanger. Flow to each SRW heat exchanger is reduced while the system remains capable of providing the required flow to the ECCS pump room air coolers.

The SW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two SW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power. Additionally, this system will also operate assuming the worst case passive failure post-RAS.

An SW subsystem is considered OPERABLE when:

a. The associated pump is OPERABLE; and

b. The associated piping, valves, heat exchangers, and instrumentation and controls required to perform the safety-related function are OPERABLE.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the SW System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the SW System and required to be OPERABLE in these MODEs.

In MODEs 5 and 6, the OPERABILITY requirements of the SW System are determined by the systems it supports.

ACTIONS

A.1

Insert 1

With one SW subsystem inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SW subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the SW subsystem could result in loss of SW System function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions of LCO 3.8.1 should be entered if the inoperable SW subsystem results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6 should be entered if an inoperable SW subsystem results in an inoperable SDC. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

Insert 11

~~B~~C.1 and ~~B~~C.2

If the SW subsystems cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the SW System flow path ensures that the proper flow paths exist for SW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This surveillance test does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the SW System components or systems may render those components inoperable but does not affect the OPERABILITY of the SW System.

BASES Insert 12

C ~~B~~.1 and C ~~B~~.2

If the Required Actions and associated Completion Times of Condition A or B are not met in MODEs 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D ~~C~~.1

If both CRETS trains are inoperable ~~in MODEs 1, 2, 3, or 4, or~~ during movement of irradiated fuel assemblies, the CRETS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, ~~LCO 3.0.3 must be entered immediately and~~ movement of irradiated fuel must be suspended immediately. This does not preclude the movement of fuel assemblies to a safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies each required CRETS train has the capability to maintain Control Room temperature ≤ 104°F for ≥ 12 hours in the recirculation mode. During this test, the backup Control Room air conditioner is to be de-energized. This SR consists of a combination of testing. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 9.8.2.3, "Auxiliary Building Ventilating Systems"

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1

With one MFIV inoperable, action must be taken to restore the valve to OPERABLE status within 72 hours. Insert 2

The 72 hour Completion Time takes into account the isolation capability afforded by the MFW regulating valves, and tripping of the MFW pumps, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

B.1 and B.2

If the MFIVs cannot be restored to OPERABLE status in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR ensures the closure time for each MFIV is $\leq$ 65 seconds by manual isolation. The MFIV closure time is assumed in the accident and containment analyses.

The Frequency is in accordance with the Inservice Testing Program. The MFIVs are tested during each refueling outage in accordance with Reference 2, and sometimes during other cold shutdown periods. The Frequency demonstrates the valve closure time at least once per refueling cycle. Operating experience has shown that these components usually pass the surveillance test when performed.

BASES

category are a feedwater line break, and a SGTR event (limiting case).

The ADVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ADV lines are required to be OPERABLE to ensure that at least one ADV is OPERABLE to conduct a unit cooldown following an event in which one steam generator becomes unavailable. A closed isolation valve does not render its ADV line inoperable since operator action time to open the isolation valve is supported in the accident analysis.

Failure to meet the LCO can result in the inability to cool the unit to SDC System entry conditions following an event in which the condenser is unavailable for use with the Turbine Bypass Valves. An ADV is considered OPERABLE when it is capable of providing relief of the main steam flow, and is capable of fully opening and closing when required.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generators are being relied upon for heat removal, the ADVs are required to be OPERABLE. In MODES 5 and 6, a SGTR is not a credible event.

ACTIONS

A.1

Insert 1

With one required ADV line inoperable, action must be taken to restore the OPERABLE status within 48 hours. The 48 hour Completion Time takes into account the redundant capability afforded by the remaining OPERABLE ADV line, and a backup in the Turbine Bypass Valves and MSSVs.

B.1

With two required ADV lines inoperable, action must be taken to restore one of the ADV lines to OPERABLE status. As the isolation valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 1 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Turbine Bypass Valves and MSSVs, and the low probability of an event occurring during this period that requires the ADV lines.

Insert 2

Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features (or both) that are associated with the other train that has offsite power. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuits and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

Insert 2

Consistent with Reference 6, operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.1

The 14 day Completion Time for Required Action B.5 is based on the OPERABILITY of both opposite-unit DGs and the availability of the OC DG. The OC DG is available to power the inoperable DG bus loads in the event of a station blackout or loss-of-offsite power. It is required to administratively verify both opposite-unit DGs OPERABLE and the OC DG available within one hour and to continue this however, is no longer under the 24 hour constraint imposed while in Condition B.

Consistent with Reference 7, 24 hours is reasonable to confirm that the OPERABLE DG(s) is not affected by the same problem as the inoperable DG.

These Conditions (B.4.1 and B.4.2) do not address the availability of the OC DG.

B.5

Operation may continue in Condition B for a period that should not exceed 14 days.

Planned entry into this Required Action requires that a risk assessment be performed in accordance with a configuration risk management program (Reference 11). This ensures that a proceduralized probabilistic risk assessment-informed process is in place that assesses the overall impact of plant maintenance on plant risk prior to entering this Required Action for planned activities.

In Condition B, the remaining OPERABLE DGs, available OC DG, and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. Insert 2

In addition to utilizing Calvert Cliffs Nuclear Power Plant's processes for evaluating risk, Reference 11, Calvert Cliffs will administratively limit DG OOS time to 72 hours for elective maintenance unless the following actions are completed:

a. Weather conditions will be evaluated prior to entering the extended DG Completion Time for elective maintenance. An extended DG Completion Time will not be entered for elective maintenance purposes if official weather forecasts are predicting severe conditions (tornado or thunderstorm warnings).

power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation could correspond to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With two of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a loss of coolant accident, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria. Insert 2

Consistent with Reference 6, with the available offsite AC sources two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A or D, as applicable.

H.1 and H.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition H are modified by a Note to indicate that when Condition H is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, must be immediately entered. This allows Condition H to provide requirements for the loss of one required LCO 3.8.1.a offsite circuit and one LCO 3.8.1.b DG without regard to whether a train is de-energized. Limiting Condition for Operation 3.8.9 provides the appropriate restrictions for a de-energized train.

Insert 2

Consistent with Reference 6, operation may continue in Condition H for a period that should not exceed 12 hours.

In Condition H, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition G (loss of two required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

Insert 2

I.1

With two LCO 3.8.1.b DGs inoperable, there are no remaining standby AC sources to provide power to most of the ESF systems. With one LCO 3.8.1.c DG inoperable and the LCO 3.8.1.b DG that provides power to the CREVS and CRETS inoperable, there are no remaining standby AC sources to the CREVS and CRETS. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a short time could

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be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

Consistent with Reference 6, with both LCO 3.8.1.b DGs inoperable, or with the LCO 3.8.1.b DG that provides power to the CREVS and CRETS and the LCO 3.8.1.c DG inoperable, operation may continue for a period that should not exceed 2 hours. Insert 2

Insert 13

K ~~J~~.1 and ~~J~~.2 K

If any Required Action and associated Completion Time of Conditions A, B.2, B.3, B.4.1, B.4.2, B.5, C, E.2, E.3, E.4.1, E.4.2, E.5, F, G, H, or I are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within six hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

~~K.1~~

~~Condition K corresponds to a level of degradation in which all redundancy in LCO 3.8.1.a and LCO 3.8.1.b AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.~~

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with Reference 1, GDC 18. Periodic component tests are supplemented by extensive functional tests during refueling associated bus, are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. Loss of any DC channel does not prevent the minimum safety function from being performed (Reference 1, Chapter 8).

An OPERABLE DC channel requires the battery and one OPERABLE charger to be operating and connected to the associated DC bus(es).

APPLICABILITY

The DC sources are required to be OPERABLE in MODEs 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC sources requirement for MODEs 5 and 6 are addressed in the Bases for LCO 3.8.5.

ACTIONS

A.1

Required Action A.1 requires the inoperable battery to be replaced by the reserve battery within four hours when one DC channel is inoperable due to an inoperable battery and the reserve battery is available. The reserve battery is a qualified battery that can replace and perform the required function of any inoperable battery. The four hour Completion Time is acceptable based on the capability of the reserve battery and the time it takes to replace the inoperable battery with the reserve battery while minimizing the time in this degraded condition. Insert 2

B.1

Condition B represents one channel with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. Therefore, it is imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected channel. The 2 hour limit is consistent with the allowed time for an inoperable DC channel. Insert 2

If one of the required DC channels is inoperable for reasons other than Condition A (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC channels have the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure would, however, result in the further loss of the 125 VDC channels with attendant loss of ESF functions, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Reference 5 and reflects a reasonable time to assess unit status as a function of the inoperable DC channel and, if the DC channel is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

Insert 14

D ~~C~~.1 and ~~C~~ D.2

If the inoperable DC channel cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Reference 5.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying connected loads and the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery (2.13 V per cell average) and are consistent

BASES

from its 120 VAC bus powered by an ESF motor control center through a regulating transformer.

Required Action A.1 is modified by a Note, which states to enter the applicable conditions and Required Actions of LCO 3.8.9, when Condition A is entered with one AC vital bus de-energized. This ensures the vital bus is re-energized within two hours.

Insert 2

Required Action A.1 allows 24 ~~hours to fix the inoperable inverter and return it to service.~~ The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

Insert 15

C~~B~~.1 and C~~B~~.2

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

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This SR verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The Surveillance Frequency subsystems require the associated buses to be energized to their proper voltage.

In addition, tie breakers between redundant safety-related AC, DC, and AC vital bus distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical distribution subsystems are considered inoperable. This applies to the onsite, safety-related redundant electrical power distribution subsystems.

APPLICABILITY

The electrical distribution subsystems are required to be OPERABLE in MODEs 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and Containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical distribution subsystem requirements for MODEs 5 and 6 are covered in the Bases for LCO 3.8.10.

ACTIONS

A.1

With one or more required AC buses, load centers, motor control centers, or distribution panels, except AC vital buses, inoperable and a loss of function has not yet occurred, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within eight hours.

Insert 1

Condition A worst scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The eight hour time limit before requiring a unit shutdown in this condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train, to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

B.1

With one or more AC vital buses inoperable and a loss of Function has not yet occurred, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the AC vital bus must be restored to OPERABLE status within two hours by powering the bus from an associated inverter via DC or the non-Class 1E 120 VAC bus powered by an ~~ESF motor~~ control center through a regulating transformer. Insert 2

Condition B represents one or more AC vital buses without power; potentially both the DC source and the associated AC source are non-functioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining vital buses, and restoring power to the affected vital bus.

This two hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, which would have the Required Action Completion Times shorter than two hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The two hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

C.1

With one DC bus inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the DC bus must be restored to OPERABLE status within two hours by ~~powering the bus from the associated battery or charger.~~ Insert 2

BASES

Condition C represents one DC bus without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

This two hour limit is more conservative than Completion Times allowed for the vast majority of components which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The two hour Completion Time for DC buses is consistent with Reference 2.

Insert 16

E ~~D~~.1 and ~~D~~.2 E

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

ATTACHMENT 4

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Cross-Reference of TSTF-505 and Calvert Cliffs Nuclear Power Plant Technical Specifications

Cross-Reference of TSTF-505 and Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Completion Times	**1.3**	**1.3**		
Example 1.3-8	1.3-8	[NEW TS] 1.3-8	No	The CCNPP TS do not currently contain this example. Example to be added to the CCNPP TS consistent with TSTF-505.
Reactor Protection System (RPS) Instrumentation - Operating	**3.3.1**	**3.3.1**		
One RPS trip unit or associated instrument channel inoperable	3.3.1.A	3.3.1.A	Yes	TSTF-505 changes are incorporated.
Two RPS trip units or associated instrument channels inoperable	3.3.1.B	3.3.1.B	Yes	TSTF-505 changes are incorporated.
One RPS automatic bypass removal channel inoperable	3.3.1.D	3.3.1.D	Yes	TSTF-505 changes are incorporated.
Two RPS automatic bypass removal channels inoperable	3.3.1.E	3.3.1.E	Yes	TSTF-505 changes are incorporated.
RPS Logic and Trip Initiation	**3.3.3**	**3.3.3**		
One Matrix logic channel inoperable	3.3.3.A	3.3.3.A	Yes	TSTF-505 changes are incorporated.

**Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications**

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Engineered Safety Features Actuation System (ESFAS) Instrumentation	**3.3.4**	**3.3.4**		
Two or more Containment Spray Actuation Signal (CSAS) trip units or associated instruments inoperable	3.3.4.B	-----	No	The CCNPP TS do not contain this Condition. Therefore, a change is not proposed to the CCNPP TS.
One ESFAS trip unit or associated instrument channel inoperable	3.3.4.C	3.3.4.A	Yes	TSTF-505 changes are incorporated.
Two ESFAS trip units or associated instrument channels inoperable	3.3.4.D	3.3.4.B	Yes	TSTF-505 changes are incorporated.
One ESFAS automatic bypass removal channel inoperable	3.3.4.E	------	No	Although the CCNPP TS do contain this Condition, the CCNPP TS do not contain the additional Required Actions contained in TSTF-505, involving the restoration of the bypass removal channel or placing the affected trip unit in trip, to which a Risk Informed Completion Time (RICT) is applied. Therefore, a change is not proposed to the CCNPP TS.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Two ESFAS automatic bypass removal channels inoperable.	3.3.4.F	3.3.4.D	Yes	TSTF-505 changes are incorporated. EDITORIAL: The CCNPP TS refer to the "two automatic bypass removal channels" as the "automatic block removal feature of two sensor block modules."
ESFAS Logic and Manual Trip	**3.3.5**	**3.3.5**		
One Auxiliary Feedwater Actuation signal (AFAS) Manual Trip or Actuation Logic channel inoperable	3.3.5.A	3.3.5.A	Yes	TSTF-505 changes are incorporated.
Two AFAS Manual Trip or Actuation Logic channels inoperable	3.3.5.B	[NEW TS] 3.3.5.B	Yes	The CCNPP TS do not currently contain this Condition. CCNPP proposes to add a new Condition B (remaining Conditions renumbered) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program, consistent with TSTF-505.
One ESFAS Manual Trip or Actuation Logic channel inoperable except AFAS	3.3.5.D	3.3.5.D	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Two ESFAS Manual Trip or Actuation Logic channels inoperable except AFAS	3.3.5.E	[NEW TS] 3.3.5.E	Yes	The CCNPP TS do not currently contain this Condition. CCNPP proposes to add a new Condition E (remaining Conditions renumbered) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program, consistent with TSTF-505.
Diesel Generator (DG) – Loss of Voltage Start (LOVS)	**3.3.6**	**3.3.6**		
One DG-LOVS channel per DG [diesel generator] inoperable	3.3.6.A	3.3.6.A	Yes	TSTF-505 changes are incorporated.
Two DG-LOVS channels per DG inoperable	3.3.6.B	3.3.6.B	Yes	TSTF-505 changes are incorporated.
More than two DG-LOVS channels inoperable	3.3.6.C	3.3.6.C	Yes	TSTF-505 changes are incorporated.
RCS Loops – MODE 3	**3.4.5**	**3.4.5**		
One RCS loop inoperable	3.4.5.A	3.4.5.A	No	Mode 3 – TSTF-505 changes excluded. CCNPP is not proposing to apply the RICT Program in Mode 3.
Pressurizer	**3.4.9**	**3.4.9**		
One group of pressurizer heaters inoperable	3.4.9.B	3.4.9.B	No	TSTF-505 changes are excluded. This TS function is not modeled in the CCNPP PRA.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Two groups of pressurizer heaters inoperable	3.4.9.C	3.4.9.C	No	TSTF-505 changes are excluded. This TS function is not modeled in the CCNPP PRA.
Pressurizer Safety Valves	**3.4.10**	**3.4.10**		
One pressurizer safety valve inoperable	3.4.10.A	3.4.10.A	Yes	TSTF-505 changes are incorporated.
Pressurizer Power Operated Relief Valves (PORVs)	**3.4.11**	**3.4.11**		
One PORV inoperable and not capable of being manually cycled	3.4.11.B	3.4.11.B	Yes	TSTF-505 changes are incorporated.
One block valve inoperable	3.4.11.C	3.4.11.C	Yes	TSTF-505 changes are incorporated.
Two PORVS inoperable and not capable of being manually cycled	3.4.11.E	3.4.11.D	Yes	TSTF-505 changes are incorporated.
Two block valves inoperable	3.4.11.F	3.4.11.E	Yes	TSTF-505 changes are incorporated.
RCS Pressure Isolation Valve (PIV) Leakage	**3.4.14**	------		The CCNPP TS do not contain this TS. Therefore, a change is not proposed to the CCNPP TS.
Safety Injection Tanks (SITs)	**3.5.1**	**3.5.1**		
One SIT inoperable due to boron concentration not within limits	3.5.1.A	3.5.1.A	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	**TSTF-505 Tech Spec**	**CCNPP Tech Spec**	**Apply RICT?**	**Comments**
One SIT inoperable for reasons other than Condition A	3.5.1.B	3.5.1.B	Yes	TSTF-505 changes are incorporated.
Two or more SITs inoperable	3.5.1.C	3.5.1.D	Yes	CCNPP proposes to change the existing immediate shutdown Required Action to a 1 hour Completion Time and the option to use the RICT Program, consistent with TSTF-505.
Emergency Core Cooling Systems (ECCS) - Operating	**3.5.2**	**3.5.2**		
One LPSI [Low Pressure Safety Injection] train	3.5.2.A	-----		The CCNPP TS do not contain this Condition. Therefore, a change is not proposed to the CCNPP TS.
One or more ECCS trains inoperable	3.5.2.B	3.5.2.A	Yes	TSTF-505 changes are incorporated. In addition, CCNPP is proposing to delete the following words from CCNPP TS Condition A: "AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available." The deletion of these words is being done because the proposed CCNPP TS 3.5.2, Condition B will address the condition for "Less than 100% of the ECCS flow equivalent to a single OPERABLE train available." Therefore, it is unnecessary to also have the wording, in CCNPP TS 3.5.2, Condition A.

Cross-Reference of TSTF-505 and Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Less than 100% of the ECCS flow equivalent to a single OPERABLE train available	3.5.2.C	[NEW TS] 3.5.2.B	Yes	The CCNPP TS do not currently contain this Condition. CCNPP proposes to add a new Condition B (remaining Conditions renumbered) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program, consistent with TSTF-505.
ECCS – Shutdown	**3.5.3**	**3.5.3**		
Required HPSI [High Pressure Safety Injection] Train inoperable	3.5.3.A	3.5.3.A	No	Mode 3 – TSTF-505 changes excluded. CCNPP is not proposing to apply the RICT Program in Mode 3.
Refueling Water Tank (RWT)	**3.5.4**	**3.5.4**		
RWT boron concentration or borated water temperature not within limits	3.5.4.A	3.5.4.A	Yes	TSTF-505 changes are incorporated.
RWT inoperable for reasons other than Condition A	3.5.4.B	3.5.4.B	Yes	TSTF-505 changes are incorporated.
Containment Air Locks	**3.6.2**	**3.6.2**		
Containment air locks inoperable for reasons other than Conditions A or B	3.6.2.C	3.6.2.C	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Containment Isolation Valves	**3.6.3**	**3.6.3**		
One containment sump supply valve inoperable	3.6.3.A	-------	No	The CCNPP TS do not contain this Condition. Therefore, a change is not proposed to the CCNPP TS.
One containment isolation valve inoperable	3.6.3.B	3.6.3.A	Yes	TSTF-505 changes are incorporated.
Two containment isolation valves inoperable	3.6.3.C	3.6.3.B	Yes	TSTF-505 changes are incorporated.
One containment isolation valve inoperable (closed system)	3.6.3.D	3.6.3.C	Yes	TSTF-505 changes are incorporated.
Secondary containment bypass or purge valve leakage not within limit	3.6.3.E	--------	No	The CCNPP TS do not contain this Condition. Therefore, a change is not proposed to the CCNPP TS.
Containment purge valves not within purge valve leakage limit	3.6.3.F	-------	No	The CCNPP TS do not contain this Condition. Therefore, a change is not proposed to the CCNPP TS.
Containment Spray and Cooling Systems	**3.6.6**	**3.6.6**		
One containment spray train inoperable	3.6.6.A	3.6.6.A	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	**TSTF-505 Tech Spec**	**CCNPP Tech Spec**	**Apply RICT?**	**Comments**
One containment cooling train inoperable	3.6.6.C	3.6.6.B	Yes	TSTF-505 changes are incorporated.
One containment spray and one containment cooling train inoperable	3.6.6.D	-------	No	The CCNPP TS do not contain this Condition. Therefore, a change is not proposed to the CCNPP TS.
Two containment cooling trains inoperable	3.6.6.E	3.6.6.D	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	**TSTF-505 Tech Spec**	**CCNPP Tech Spec**	**Apply RICT?**	**Comments**
Two containment spray trains or any combination of three or more containment spray and cooling trains inoperable	3.6.6.F	3.6.6.C 3.6.6.F	Yes	NUREG-1432 combines the Conditions for two containment spray trains or any combination of three or more containment spray and cooling trains inoperable into the one Condition G. TSTF-505 converts this Condition into a new Condition F with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program. CCNPP TS has two separate Conditions instead of one: Condition C for two containment spray trains inoperable and Condition F for any combination of three or more trains inoperable. CCNPP TS 3.6.6, Condition C already contains a restoration action with a Completion Time of 24 hours. Therefore, CCNPP proposes to apply a RICT to the existing CCNPP TS 3.6.6, Condition C, consistent with other similar TSTF-505 changes. For CCNPP TS 3.6.6, Condition F, the Required Action is an immediate shutdown. Consistent with the changes reflected in TSTF-505 TS 3.6.6, Condition F, CCNPP proposes to change the immediate shutdown Required Action in Condition F to adopt the 1 hour Completion Time and the option to use the RICT Program.
Hydrogen Mixing System (HMS)	**3.6.9**	**-------**		The CCNPP TS do not contain this TS. Therefore, a change is not proposed to the CCNPP TS.

Cross-Reference of TSTF-505 and Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Main Steam Isolation Valves (MSIVs)	**3.7.2**	**3.7.2**		
One MSIV inoperable in Mode 1	3.7.2.A	3.7.2.A	Yes	TSTF-505 changes are incorporated.
Two or more MSIVs inoperable in Mode 1	3.7.2.C	[NEW TS] 3.7.2.C	Yes	The CCNPP TS do not currently contain this Condition. CCNPP proposes to add a new Condition C (remaining Conditions renumbered) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program, consistent with TSTF-505.
Main Feedwater Isolation Valves (MFIVs)	**3.7.3**	**3.7.15**		
One or more MFIVs inoperable	3.7.3.A	3.7.15.A	Yes	The TSTF-505 has a comment stating that Conditions A and B do not specify a restoration action and Condition C is a default Condition, thus the LCO conditions were excluded. However, the corresponding CCNPP TS 3.7.15, Condition A for one or more MFIVs inoperable does contain a restoration action with a Completion Time of 72 hours. Therefore, CCNPP proposes to apply a RICT to the existing CCNPP TS 3.7.15, Condition A. This is acceptable because the TSTF states that there may also be plant-specific TS to which changes of the type presented in the TSTF may be applied.

Cross-Reference of TSTF-505 and Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Automatic Dump Valves (ADVs)	**3.7.4**	**3.7.18**		
One required ADV line inoperable	3.7.4.A	3.7.18.A	Yes	TSTF-505 changes are incorporated.
Two required ADV lines inoperable	3.7.4.B	3.7.18.B	Yes	TSTF-505 changes are incorporated.
Auxiliary Feedwater (AFW) System	**3.7.5**	**3.7.3**		
One steam supply to turbine driven AFW pump inoperable	3.7.5.A	3.7.3.A	Yes	TSTF-505 changes are incorporated. EDITORIAL: CCNPP TS 3.7.3.A states: "One steam-driven AFW pump inoperable."
One motor-driven AFW pump inoperable	-------	3.7.3.B	Yes	This is a plant-specific Condition with a restoration action and a Completion Time of 7 days. CCNPP proposes to apply a RICT to the existing CCNPP TS 3.7.3, Condition B. This is acceptable because the TSTF states that there may also be plant-specific TS to which changes of the type presented in the TSTF may be applied.
Two AFW pumps inoperable	-------	3.7.3.C	Yes	This is a plant-specific Condition with a restoration action and a Completion Time of 72 hours. CCNPP proposes to apply a RICT to the existing CCNPP TS 3.7.3, Condition C. This is acceptable because the TSTF states that there may also be plant-specific TS to which changes of the type presented in the TSTF may be applied.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
One AFW train inoperable	3.7.5.B	3.7.3.D	Yes	TSTF-505 changes are incorporated.
Two AFW trains inoperable	3.7.5.C	-----	No	NUREG-1432 TS 3.7.5 is based on plants that have a three-train AFW system. CCNPP only has two trains of AFW. Therefore, since two AFW trains inoperable represents a loss of safety function for CCNPP, CCNPP does not propose to incorporate the changes reflected in TSTF-505 TS 3.7.5, Condition C.
Condensate Storage Tank (CST)	**3.7.6**	**3.7.4**		
CST inoperable	3.7.6.A	3.7.4.A	Yes	TSTF-505 changes are incorporated.
Component Cooling Water (CCW) System	**3.7.7**	**3.7.5**		**EDITROIAL: Component Cooling (CC) System for CCNPP**
One CCW train inoperable	3.7.7.A	3.7.5.A	Yes	TSTF-505 changes are incorporated. EDITORIAL: "CC Loops" inoperable for CCNPP.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Two CCW trains inoperable	3.7.7.B	[NEW TS] 3.7.5.B	Yes	The CCNPP TS do not currently contain this Condition. CCNPP proposes to add a new Condition B (*remaining Conditions renumbered*) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program, consistent with TSTF-505. Note: An error was identified in TSTF-505 in that the new TS 3.7.7, Condition B did not contain the typical Note that the Condition is not applicable when the second CCW train is intentionally made inoperable. This Note is proposed to be included in the new CCNPP TS 3.7.5, Condition B.
Service Water Systems (SWS)	**3.7.8**	**3.7.6**		**EDITORIAL: Service Water (SRW) System for CCNPP**
One SRW heat exchanger inoperable	--------	3.7.6.A	Yes	This is a plant-specific Condition with a restoration action and a Completion Time of 7 days. CCNPP proposes to apply a RICT to the existing CCNPP TS 3.7.6, Condition A. This is acceptable because the TSTF states that there may also be plant-specific TS to which changes of the type presented in the TSTF may be applied.
One SWS train inoperable	3.7.8.A	3.7.6.B	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Two SWS trains inoperable	3.7.8.B	[NEW TS] 3.7.6.C	Yes	The CCNPP TS do not currently contain this Condition. CCNPP proposes to add a new Condition C (remaining Conditions renumbered) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program, consistent with TSTF-505.
Saltwater (SW) System	**-----------**	**3.7.7**		**Note: CCNPP's Saltwater System performs functions similar to the Ultimate Heat Sink reflected in NUREG-1432 TS 3.7.9; however, the CCNPP Saltwater TS 3.7.7 was modeled after the CCNPP Service Water System TS 3.7.6 and NUREG-1432 TS 3.7.8.**
One SW subsystem inoperable	---------	3.7.7.A	Yes	This is a plant-specific Condition with a restoration action and a Completion Time of 7 days. CCNPP proposes to apply a RICT to the existing CCNPP TS 3.7.7, Condition A. This is acceptable because the TSTF states that there may also be plant-specific TS to which changes of the type presented in the TSTF may be applied.
Two SW subsystems inoperable	---------	[NEW TS] 3.7.7.B	Yes	CCNPP proposes to add a new Condition B (remaining Conditions renumbered) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program, consistent with other similar TSTF-505 changes.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Ultimate Heat Sink (UHS)	**3.7.9**	**------**		The CCNPP TS do not contain this TS. Therefore, a change is not proposed to the CCNPP TS.
Essential Chilled Water (ECW)	**3.7.10**	**------**		The CCNPP TS do not contain this TS. Therefore, a change is not proposed to the CCNPP TS.
Control Room Emergency Air Temperature Control System (CREATCS)	**3.7.12**	**3.7.9**		**EDITORIAL: Control Room Emergency Temperature System (CRETS) for CCNPP**
Two CRETS trains during Modes 1, 2, 3 and 4	3.7.12.B	[NEW TS] 3.7.9.B	Yes	NUREG-1432 TS 3.7.12, Condition E requires an immediate shutdown for two CREATCS trains inoperable in Modes 1, 2, 3 and 4. TSTF-505 converts this Condition into a new Condition B with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program. CCNPP TS 3.7.9, Condition C currently combines two conditions with separate Required Actions and CTs; one for two CRETS trains inoperable during Modes 1, 2, 3 and 4 (which requires an immediate shutdown similar to NUREG-1432 TS 3.7.12, Condition E), and the other for two CRETS trains inoperable during movement of irradiated fuel assemblies (which requires immediate suspension of movement of irradiated fuel assemblies similar to NUREG-1432 TS 3.7.12, Condition D). Consistent with TSTF-505, CCNPP proposes to delete the current portion of TS 3.7.9, Condition C

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
				pertaining to two CRETS trains inoperable during Modes 1, 2, 3 and 4, and convert it to a new Condition B (remaining Conditions renumbered) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program. Because of the changes to CCNPP TS 3.7.9, Condition C described above, the resultant CCNPP TS 3.7.9, Condition C (renumbered to D) will remain applicable to the condition of two CRETS trains inoperable during the movement of irradiated fuel assemblies, and will require immediate suspension of the movement of irradiated fuel assemblies similar to NUREG-1432 TS 3.7.12, Condition D (renumbered E by TSTF-505).
AC Sources - Operating	**3.8.1**	**3.8.1**		
One offsite circuit inoperable	3.8.1.A	3.8.1.A	Yes	TSTF-505 changes are incorporated.
One DG [diesel generator] inoperable	3.8.1.B	3.8.1.B	Yes	TSTF-505 changes are incorporated.
Two offsite circuits inoperable	3.8.1.C	3.8.1.G	Yes	TSTF-505 changes are incorporated.
One offsite circuit and one DG inoperable	3.8.1.D	3.8.1.H	Yes	TSTF-505 changes are incorporated.
Two DGs inoperable	3.8.1.E	3.8.1.I	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
One automatic load sequencer inoperable	3.8.1.F	------	No	The CCNPP TS do not contain this Condition. Therefore, a change is not proposed to the CCNPP TS.
Three or more AC sources inoperable	3.8.1.G	3.8.1.J	Yes	TSTF-505 changes are incorporated.
DC Sources - Operating	**3.8.4**	**3.8.4**		
One or two battery chargers on one train inoperable	3.8.4.A	--------	No	The CCNPP TS do not contain this Condition. Therefore, a change is not proposed to the CCNPP TS.
One or two batteries on one train inoperable	3.8.4.B	3.8.4.A	Yes	TSTF-505 changes are incorporated. EDITORIAL: CCNPP TS 3.8.4 refers to one DC channel inoperable due to an inoperable battery rather than one battery inoperable on one train.
One DC electrical power subsystems inoperable	3.8.4.C	3.8.4.B	Yes	TSTF-505 changes are incorporated. EDITORIAL: CCNPP TS 3.8.4 refers to DC "channels" rather than "electrical power subsystems."

**Cross-Reference of TSTF-505 and
Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications**

Tech Spec Description	TSTF-505 Tech Spec	CCNPP Tech Spec	Apply RICT?	Comments
Two DC electrical power subsystems inoperable	3.8.4.D	[NEW TS] 3.8.4.C	Yes	TSTF-505 changes are incorporated Note: NUREG-1432 TS 3.8.4 is based on a plant with two DC electrical power subsystems with each subsystem consisting of two 125 VDC batteries. CCNPP TS 3.8.4 is based on four DC channels with each channel consisting of one 125 VDC battery.
Inverters - Operating	**3.8.7**	**3.8.7**		
One inverter inoperable	3.8.7.A	3.8.7.A	Yes	TSTF-505 changes are incorporated.
Two or more inverters inoperable	3.8.7.B	[NEW TS] 3.8.7.B	Yes	The CCNPP TS do not currently contain this action. CCNPP proposes to add a new Condition B (remaining Conditions renumbered) with a new Required Action with a 1 hour Completion Time and the option to use the RICT Program, consistent with TSTF-505.
Distribution Systems - Operating	**3.8.9**	**3.8.9**		
One or more AC electrical distribution subsystems inoperable	3.8.9.A	3.8.9.A	Yes	TSTF-505 changes are incorporated.
One or more AC vital buses inoperable	3.8.9.B	3.8.9.B	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specifications

Tech Spec Description	**TSTF-505 Tech Spec**	**CCNPP Tech Spec**	**Apply RICT?**	**Comments**
One or more DC electrical power distribution subsystems inoperable	3.8.9.C	3.8.9.C	Yes	TSTF-505 changes are incorporated. EDITORIAL: CCNPP is limited to only one DC electrical power distribution subsystem inoperable.
Two or more electrical power distribution subsystems inoperable that result in a loss of safety function	3.8.9.D	3.8.9.D	Yes	TSTF-505 changes are incorporated.
Programs and Manuals	**5.5**	**5.5**		
Programs and Manuals	5.5.18	[NEW TS] 5.5.18	No	The CCNPP TS do not currently contain this program. The new RICT Program will be added to the CCNPP TS consistent with TSTF-505.

ENCLOSURE 1

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

<u>List of Revised Required Actions to Corresponding PRA Functions</u>

List of Revised Required Actions to Corresponding PRA Functions

1. Introduction

Section 4.0, Item 2 of the NRC Final Safety Evaluation (Reference 1 of this Enclosure) for NEI 06-09, Revision 0-A, Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, (Reference 2) identifies the following needed content:

- The license amendment request (LAR) will provide identification of the TS Limiting Conditions for Operation (LCOs) and action requirements to which the RMTS will apply.
- The LAR will provide a comparison of the TS functions to the PRA modeled functions of the structures, systems, and components (SSCs) subject to those LCO actions.
- The comparison should justify that the scope of the PRA model, including applicable success criteria such as number of SSCs required, flow rate, etc., are consistent with licensing basis assumptions (i.e., 50.46 ECCS flowrates)for each of the TS requirements, or an appropriate disposition or programmatic restriction will be provided.

This enclosure provides confirmation that the Calvert Cliffs Nuclear Power Plant (CCNPP) PRA models include the necessary scope of SSCs and their functions to address each proposed application of the Risk-Informed Completion Time (RICT) Program to the proposed scope TS LCO Conditions, and provides the information requested for Section 4.0, Item 2 of the NRC Final Safety Evaluation. The scope of the comparison includes each of the TS LCO conditions and associated required actions within the scope of the RICT Program. The CCNPP PRA model has the capability to model directly or through use of a bounding surrogate the risk impact of entering each of the TS LCOs in the scope of the RICT Program.

Table E1-1 below lists each TS LCO Condition to which the RICT Program is proposed to be applied and documents the following information regarding the TSs with the associated safety analyses, the analogous PRA functions and the results of the comparison:

- Column "Proposed TS LCO Condition": Lists all of the LCOs and condition statements within the scope of the RICT Program.
- Column "SSCs Covered by TS LCO Condition": The SSCs addressed by each action requirement.
- Column "SSCs Modeled in PRA": Indicates whether the SSCs addressed by the TS LCO Condition are included in the PRA.
- Column "Function Covered by TS LCO Condition": A summary of the required functions from the design basis analyses.
- Column "Design Success Criteria": A summary of the success criteria from the design basis analyses.
- Column "PRA Success Criteria": The function success criteria modeled in the PRA.
- Column "Comments": Provides the justification or resolution to address any inconsistencies between the TS and PRA functions regarding the scope of SSCs and the success criteria. Where the PRA scope of SSCs is not consistent with the TS, additional information is provided to describe how the LCO condition can be evaluated using appropriate surrogate events. Differences in the success criteria for TS functions are addressed to demonstrate the PRA criteria provide a realistic estimate of the risk of the TS condition as required by NEI 06-09 Revision 0-A.

List of Revised Required Actions to Corresponding PRA Functions

The corresponding SSCs for each TS LCO and the associated TS functions are identified and compared to the PRA. This description also includes the design success criteria and the applicable PRA success criteria. Any differences between the scope or success criteria are described in the table. Scope differences are justified by identifying appropriate surrogate events which permit a risk evaluation to be completed using the CRMP tool for the RICT program. Differences in success criteria typically arise due to the requirement in the PRA standard to make PRAs realistic rather than bounding, whereas design basis criteria are necessarily conservative and bounding. The use of realistic success criteria is necessary to conform to capability Category II of the PRA standard as required by NEI 06-09 Revision 0-A.

For those TS line items (and SSCs) at CCNPP that represent loss of TS-specified safety function identified in Table E1-1, the success criteria assumed in the PRA model are consistent with the design basis criteria. Furthermore, there are no alternative SSCs (to those referenced in the TS) credited in the PRA model - used to fulfill a specified safety function for purposes of a PRA functional determination - where the SSCs referenced in the TS would be unavailable.

Examples of calculated RICT are provided in Table E1-2 for each individual condition to which the RICT applies (assuming no other SSCs modeled in the PRA are unavailable). The RICTs presented in the table are based on a Unit 1 model calculation. Due to the close similarity between the Unit 1 and Unit 2 models, the Unit 1 RICTs are considered adequate examples for the Unit 2 RICTs as well. Following 4b implementation, the actual RICT values will be calculated on a unit-specific basis, using the actual plant configuration and the current revision of the PRA model representing the as-built, as-operated condition of the plant, as required by NEI 06-09 and the NRC safety evaluation, and may differ from the RICTs presented.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.1.A One RPS bistable trip unit or measurement channel inoperable	Four RPS bistable trip units and measurement channels	YES	Each trip unit receives processed signals from its respective RPS measurement channel and provides output (reactor trip) signals in the form of relay contacts that make up the coincidence logic matrices.	Three of four RPS channels.	Minimum of 2 channels functional.	The PRA model explicitly models all 4 channels and allows each channel to be in service, in trip, or in bypass. Any two channels are required for the logic to actuate and this is reflected in the PRA model.
3.3.1.B Two RPS bistable trip units or measurement channels inoperable	See LCO Condition 3.3.1A					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.3.1.D One RPS automatic bypass removal feature channel inoperable	RPS bypass removal channels	Not Explicitly	Restores RPS input when reactor conditions indicate that the input is appropriate.	All RPS applicable automatic bypass removal features.	Per associated RPS input.	If power ascension occurred with auto bypass removal inoperable and manual bypass removal did not occur, that is equivalent to operating without the protection of the bypassed features and is equivalent to the loss of an RPS input LCO condition and would be assessed as such in the RICT calculation.
3.3.1.E Two RPS automatic bypass removal feature channels inoperable	See LCO Condition 3.3.1.D					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.3.A One RPS Matrix logic channel inoperable	RPS Matrix Logic (Six Channels)	YES	Reactor Trip Initiation	One of 6 matrix logic channels	One of 6 matrix logic channels	One logic matrix will trigger the plant trip.
3.3.4.A One ESFAS module or measurement channel inoperable	Containment pressure (SIAS,CIS,CSAS), Pressurizer pressure (SIAS), Pen room pressure (for CVCIS), SG Pressure (SGIS), RWT level (RAS), SG Level (AFAS), SG Pressure differential (AFAS Block)	YES	CSAS signals are generated for a valid containment overpressure condition. SGIS logic module provides an actuation signal to auxiliary relays. CIS logic module provides an actuation signal for a valid high containment pressure signal. RAS signals are generated for a valid RWT low level condition. SIAS signals are generated for either a valid low pressurizer pressure, or a high containment pressure condition.	Three of 4 modules or measurement channels per actuation system.	Two of 4 channels including instrumentation for each actuation system.	Two sensor channels are required for actuation of the logic.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.4.B Two ESFAS modules or measurement channels inoperable	See LCO Condition 3.3.4.A					
3.3.4.D Two ESFAS block removal features inoperable	Pressurizer pressure (SIAS), SG Pressure (SGIS)	Not explicitly	Bypass certain inputs when reactor conditions do not require them.	All ESFAS automatic block removal features for each input	Per associated ESFAS input.	If power ascension occurred with auto bypass removal inoperable and manual bypass removal did not occur, that is equivalent to operating without the protection of the bypassed features and is equivalent to the loss of an ESFAS input LCO condition and would be assessed as such in the RICT calculation.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.3.5.A One AFAS actuation channel (manual or automatic) inoperable	Two ASFAS actuation channels (automatic and manual).	Yes	Actuation of Auxiliary Feedwater.	One channel.	One channel.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.3.5.B Two AFAS channels (loss of safety function) inoperable	See LCO Condition 3.3.5.A					
3.3.5.D One ESFAS actuation channel except AFAS inoperable	See above	Yes	Actuation of Engineered Safety Features Actuation Systems	One channel.	One channel.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.3.5.E Two ESFAS actuation channels (loss of safety function) except AFAS inoperable	See 3.3.5.D above					
3.3.6.A One DG-LOVS channel inoperable	Sustained undervoltage (SUR), Transient undervoltage (TUR), or Loss of voltage (LOV) sensors on safety related 4kV buses	Not explicitly	Diesel Generator - Loss of Voltage Start as well as 4kV Bus load shedding and initiating and sequencing.	Two of four sensor modules and measurement channels per DG for the Loss of Voltage Function, two of four sensor modules and measurement channels per DG for the Transient Degraded Voltage Function, and two of four sensor modules and measurement channels per DG for the Steady State Degraded Voltage Function..	Two of four channels	.The LOV modules are modeled using the associated undervoltage relay and so can be directly evaluated using the CRMP tool. The SUR and TUR modules are not explicitly modeled and would also be addressed using the associated undervoltage relay. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.3.6.B Two DG-LOVS channels inoperable	See LCO Condition 3.3.6.A					
3.3.6.C More than two DG-LOVS channels inoperable	See LCO Condition 3.3.6.A					
3.4.10.A One pressurizer safety valve inoperable	Two Pressurizer Safety valves	Yes	Prevent RCS pressure from exceeding limit	Two pressurizer safety valves.	If the PORVs are demanded to open and fail to open given a successful reactor trip, one of two PSVs must open. If the reactor fails to trip (ATWS) regardless of PORV status both PSVs must open.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The capacity of the pressurizer safety valves is such that in a non-ATWS event a single pressurizer safety valve will prevent overpressurization of the RCS.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.4.11.B One PORV inoperable	Two PORVS per Unit	Yes	RCS depressurization, once through core cooling (feed and bleed).	Two PORVs.	Both PORVs open for RCS depressurization for core damage sequences. For LERF sequences, one PORV is sufficient for RCS depressurization.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria for CDF.
3.4.11.C One block valve inoperable	2 PORV block valves per Unit	YES	Isolate associated PORV	Two PORV Block valves closable.	Two PORV block valves closable.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.4.11.D Two PORVS inoperable	See LCO Condition 3.4.11.B					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.4.11.E Two block valves inoperable	See LCO Condition 3.4.11.C					
3.5.1.A One SIT (boron concentration) inoperable	4 Safety Injection Tanks (SITs)	Not explicitly	Provide emergency borated water source	Three of four SITs	Three of four SITs	Boron concentration in the SIT is not modeled in the PRA however this would be treated as a loss of that SIT and would be assessed as such in the RICT calculation.
3.5.1.B One SIT inoperable	4 Safety Injection Tanks (SITs)	Yes	Provide emergency borated water source	Three of four SITs	Three of four SITs	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.5.1.D Two or more SITs inoperable	See LCO Condition 3.5.1.B					
3.5.2.A One ECCS train inoperable	2 ECCS trains (HPSI and LPSI pump in each train)	Yes	Emergency make up to the RCS via injection from the RWT to the cold legs, and recirculation from the containment sump.	1 of 2 ECCS trains (HPSI and LPSI pump in each train)	1 of 2 ECCS trains Includes credit for swing HPSI pump manually initiated.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP.
3.5.2.B Less than 100% flow of ECCS train (loss of safety function)	See LCO Condition 3.5.2.A					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.5.4.A RWT boron or temp inoperable	Refueling Water Tank (RWT)	Not Explicitly	Supply borated water to ECCS and Containment Spray during LOCA injection phase for: negative reactivity for reactor shut down and core and containment cooling and containment depressurization	The RWT Boron concentration and temperature within limits.	The RWT Boron concentration and temperature within limits.	The PRA does not explicitly model the impact of out of limit boron concentration or temperature, but conservatively these can be addressed for the RICT Program by assuming the RWT is unavailable. Therefore, this LCO condition can be evaluated using the CRMP tool.
3.5.4.B RWT inoperable	RWT	Yes	Refueling Water Tank (RWT)	360,000 gallons of borated water.	360,000 gallons of borated water.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.6.2.C Containment air lock inoperable other than a single door or interlock inoperable	Containment Airlocks	No	Containment integrity	One of two containment air lock doors closed.	One of two containment air lock doors closed.	The success criteria in the PRA are consistent with the design basis criteria. The containment airlock is not modeled in the PRA. This type of failure will be conservatively analyzed as an early containment failure.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.6.3.A One containment isolation valve/ open system inoperable	2 active or passive isolation devices on each fluid penetration line	Yes	Containment boundary and minimization of RCS inventory loss	One of 2 isolation devices per penetration.	One of 2 isolation devices per penetration.	Not all containment isolation devices are modeled to support LERF analysis, as some are screened due to release path size or other considerations. Those SSCs that are modeled consistent with the TS scope can be directly evaluated using the CRMP tool. Those SSCs that are not modeled consistent with the TS scope can be conservatively analyzed by assessing a limiting case isolation device that is modeled. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.6.3.B Two containment isolation valves/open system inoperable	See LCO Condition 3.6.3.A					
3.6.3.C One containment isolation valve/closed system inoperable	See LCO Condition 3.6.3.A					
3.6.6.A One spray train inoperable	2 containment spray trains	Yes	Provide Containment atmosphere cooling and limit post-accident pressure increase and iodine removal	One of two containment spray trains.	One of two containment spray trains.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.6.6.B One cont. cooling train inoperable	Four containment air coolers comprising two trains	Yes	Provide Containment atmosphere cooling	Two of four CACs.	Two of four CACs	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.6.6.C Two spray trains inoperable	See LCO Condition 3.6.6.A.					
3.6.6.D Two containment cooling trains inoperable	See LCO Condition 3.6.6.B					
3.6.6.F Three trains inoperable (containment spray or containment cooling)	See LCO Condition 3.6.6.A & 3.6.6.B.					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.7.2.A One MSIV inoperable	Main Steam Isolation Valves (MSIVs)	Yes	Isolate the main steam lines	Both MSIVs close	Both MSIVs close	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.7.2.C Two MSIVs inoperable	See LCO Condition 3.7.2.A					
3.7.3.A One steam driven AFW pump inoperable	Two steam driven AFW pumps	Yes	Supply feedwater to steam generators to remove RCS decay heat	One of three AFW pumps (motor or steam driven)	One of three AFW pumps (motor or steam driven).	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.7.3.B One motor driven AFW pump inoperable	One motor driven AFW pump	Yes	Supply feedwater to steam generators to remove RCS decay heat	One of three AFW pumps (motor or steam driven)	One of three AFW pumps (motor or steam driven).	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.7.3.C Two AFW pumps inoperable	See LCO Condition 3.7.3.A & 3.7.3.B					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	***Comments***
3.7.3.D One AFW train other than pumps inoperable	AFW valves and flowpath.	Yes	Supply feedwater to steam generators to remove RCS decay heat	One AFW train.	One of three AFW pumps injecting to 2 of 4 flow paths if not re-throttled, one of 4 flow paths if re-throttled.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are generally consistent with the design basis criteria. The single flow path provides the required cooling flow to the steam generator by *redirecting the flow* from the other steam generator.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.7.4.A CST inoperable	12 CST	Yes	Safety grade source of water for steam generators for removing heat from the RCS.	12 CST shall be operable with >150K gallons per unit	CST 12	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.7.5.A One CC loop inoperable	Two CC loops comprised of 1 pump and head tank with associated valves, heat exchanger, instrumentation and controls.	Yes	Heat sink for removing process and operating heat from safety related components	1 of 2 CC loops	1 of 2 CC heat exchangers, 1 of 3 pumps	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.7.5.B Two CC Loops inoperable	See LCO Condition 3.7.5.A					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.7.6.A One SRW Heat exchanger inoperable	4 SRW heat exchangers	Yes	Heat sink for removing process and operating heat from safety related components	One of two SRW heat exchangers per subsystem.	One of two SRW heat exchangers per subsystem	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.7.6.B One SRW subsystem inoperable	Three SRW pumps and 2 flow paths	YES	Heat sink for removing process and operating heat from safety related components	One of two SRW subsystems.	One of two SRW subsystems.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.7.6.C Two SRW subsystems inoperable	See LCO Condition 3.7.6.B					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.7.7.A One SW subsystem inoperable	Two trains of SW	Yes	Heat sink for removing process and operating heat from safety related components	One of two SW subsystems	One of two SW subsystems.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.7.7.B Two SW subsystems inoperable	See LCO Condition 3.7.7.A					
3.7.9.B Two CRETS trains inoperable	11 and 12 CR HVAC	Yes	Maintain control room temperature within limits	One of two CRETS trains	One of two CRETS trains	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.7.15.A One or more MFIVs inoperable	Two main feedwater isolation valves	Yes	Isolation of main feedwater lines	Both valves close	Both valves close	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria
3.7.18.A One required ADV inoperable	Two ADVs	Yes	Cool unit to shutdown cooling entry conditions when main condenser is unavailable	1 ADV	1 ADV	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.7.18.B Two required ADVs inoperable	See LCO Condition 3.7.18.A.					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.8.1.A One offsite power source inoperable	Two qualified circuits between the offsite transmission network and the onsite 1E AC Electrical Power Distribution System.	Yes	Provide power from offsite transmission network to onsite Class one buses.	One qualified circuit between the offsite transmission network and the onsite 1E AC Electrical Power Distribution System.	One qualified circuit between the offsite transmission network and the onsite 1E AC Electrical Power Distribution System.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.8.1.B One DG inoperable	Four EDGs. Two per unit.	Yes	Provide power to safety related buses when offsite power to them is lost.	One of two diesel generator (DGs) capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System	One of two diesel generator (DGs) capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.1.G Two offsite power sources inoperable Or Offsite source and EDG to CREVS/CRETS power supply inoperable	See LCO Condition 3.8.1.A					
3.8.1.H One offsite power source and one DG inoperable	See LCO Condition 3.8.1.A & 3.8.1.B					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.8.1.I Two dedicated unit DGs inoperable or Dedicated unit DG and opposite unit EDG supplying CREVS/CRETS inoperable.	Two EDGs per unit *or* 1A and 2B EDGs	Yes	Provide power to safety related buses when offsite power to them is lost and Emergency power supply to control room emergency ventilation systems.	One diesel generator (DGs) capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System for dedicated unit EDGs. One DG capable of supplying power to the CREVS and CRETS.	Same, except some portions of CREVS are not included in the PRA model.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.1.J Three or more offsite and DGs inoperable	See LCO Condition 3.8.1.A, 3.8.1.B & 3.8.1.I					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.4.A One battery inoperable and reserve battery available	DC batteries	Yes	Ensure availability of required DC power to shut down the reactor and maintain it in a safe condition	Primary or reserve battery for each channel. Three of four channels.	Primary or reserve battery for each channel. Three of four channels.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.4.B One DC channel inoperable	DC batteries, battery chargers, cabling and controls	Yes	Ensure availability of required DC power to shut down the reactor and maintain it in a safe condition	One of two chargers for each channel. Primary or reserve battery for each channel. Three of four channels.	One of two chargers for each channel. Primary or reserve battery for each channel. Three of four channels.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.4.C Four DC channels inoperable	See LCO Condition 3.8.4.B					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.7.A One inverter inoperable	Four inverters per unit.	Yes	Provide AC power to vital buses	Three of four inverters	Three of four inverters	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.7.B Two or more inverters inoperable	See LCO Condition 3.8.7.A					
3.8.9.A One AC distribution subsystem inoperable	Two divisions	Yes	Provide power to safety related equipment.	One of two AC distribution systems	One of two AC distribution systems	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions

Proposed TS LCO Condition	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.9.B One or more AC vital subsystems inoperable	Four vital AC buses	Yes	Provide AC power to RPS and ESFAS	Two of four vital AC buses.	Two of four vital AC buses	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.9.C One DC distribution subsystem inoperable	Four DC distribution trains	Yes	Ensure availability of required DC power to shut down the reactor and maintain it in a safe condition.	Three of four DC electrical power distribution subsystems	Three of four DC electrical power distribution subsystems	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.8.9.D Two or more distribution subsystems (AC or DC) inoperable	See LCO Condition 3.8.9.A and 3.8.9.C.					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-2: In Scope TS/LCO Conditions RICT Estimate	
Proposed TS/LCO Condition	**RICT Estimate[1,2]**
3.3.1.A One RPS bistable trip unit or measurement channel inoperable	30 days
3.3.1.B Two RPS bistable trip units or measurement channels inoperable	30 days
3.3.1.D One RPS automatic bypass removal feature channel inoperable	30 days
3.3.1.E Two RPS automatic bypass removal feature channels inoperable	30 days
3.3.3.A One RPS Matrix logic channel inoperable	30 days
3.3.4.A One ESFAS module or measurement channel inoperable	30 days
3.3.4.B Two ESFAS modules or measurement channels inoperable	2 days[3]
3.3.4.D Two ESFAS block removal features inoperable	30 days
3.3.5.A One AFAS actuation channel (manual or automatic) inoperable	22 days
3.3.5.B Two AFAS channels inoperable	4 days[4]
3.3.5.D One ESFAS actuation channel except AFAS inoperable	28 days
3.3.5.E Two ESFAS actuation channels except AFAS inoperable	11 hours
3.3.6.A One DG-LOVS channel inoperable	30 days
3.3.6.B Two DG-LOVS channels inoperable	30 days
3.3.6.C More than two DG-LOVS channels inoperable	30 days
3.4.10.A One pressurizer safety valve inoperable	30 days
3.4.11.B One PORV inoperable	30 days
3.4.11.C One block valve inoperable	30 days
3.4.11.D Two PORVS inoperable	30 days
3.4.11.E Two block valves inoperable	30 days
3.5.1.A One SIT (boron concentration) inoperable	30 days
3.5.1.B One SIT inoperable	30 days
3.5.1.D Two or more SITs inoperable	30 days
3.5.2.A One ECCS train inoperable	30 days
3.5.2.B Less than 100% flow of ECCS train	16 hours
3.5.4.A RWT boron or temp inoperable	16 hours
3.5.4.B RWT inoperable	16 hours

List of Revised Required Actions to Corresponding PRA Functions

Table E1-2: In Scope TS/LCO Conditions RICT Estimate	
Proposed TS/LCO Condition	**RICT Estimate[1,2]**
3.6.2.C Containment air lock inoperable other than a single door or interlock inoperable	5 days[4]
3.6.3.A One containment isolation valve/ open system inoperable	30 days
3.6.3.B Two containment isolation valves/open system inoperable	30 days
3.6.3.C One containment isolation valve/closed system inoperable	30 days
3.6.6.A One spray train inoperable	30 days
3.6.6.B One cont. cooling train inoperable	30 days
3.6.6.C Two spray trains inoperable	14 days
3.6.6.D Two containment cooling trains inoperable	30 days
3.6.6.F Three trains inoperable (containment spray or containment cooling)	16 hours
3.7.2.A One MSIV inoperable	15 days[4]
3.7.2.C Two MSIVs inoperable	3 days[4]
3.7.3.A One steam driven AFW pump inoperable	30 days
3.7.3.B One motor driven AFW pump inoperable	18 days
3.7.3.C Two AFW pumps inoperable	3 days[3]
3.7.3.D One AFW train other than pumps inoperable	3 days[3]
3.7.4.A CST inoperable	7 days[3]
3.7.5.A One CC loop inoperable	30 days
3.7.5.B Two CC Loops inoperable	18 hours
3.7.6.A One SRW Heat exchanger inoperable	30 days
3.7.6.B One SRW subsystem inoperable	30 days
3.7.6.C Two SRW subsystems inoperable	30 days[5]
3.7.7.A One SW subsystem inoperable	21 days
3.7.7.B Two SW subsystems inoperable	9 hours[5]
3.7.9.B Two CRETS trains inoperable	30 days
3.7.15.A One or more MFIVs inoperable	24 days[4]
3.7.18.A One required ADV inoperable	30 days
3.7.18.B Two required ADVs inoperable	23 days[4]

List of Revised Required Actions to Corresponding PRA Functions

Table E1-2: In Scope TS/LCO Conditions RICT Estimate	
Proposed TS/LCO Condition	**RICT Estimate[1,2]**
3.8.1.A One offsite power source inoperable	30 days
3.8.1.B One DG inoperable	30 days
3.8.1.G Two offsite power sources inoperable or Offsite source and EDG to CREV/CRETs power supply inoperable	16 days
3.8.1.H One offsite power source and one DG inoperable	10 days
3.8.1.I Two dedicated unit DGs inoperable OR Dedicated unit DG and opposite unit EDG supplying CREVS/CRETS inoperable.	12 days
3.8.1.J Three or more offsite and DGs inoperable	2 days
3.8.4.A One battery inoperable and reserve battery available	2 days
3.8.4.B One DC channel inoperable	15 hours[4]
3.8.4.C Four DC channels inoperable	1 hour[3]
3.8.7.A One inverter inoperable	12 days[4]
3.8.7.B Two or more inverters inoperable	1 day
3.8.9.A One AC distribution subsystem inoperable	4 days
3.8.9.B One or more AC vital subsystems inoperable	12 days[4]
3.8.9.C One DC distribution subsystem inoperable	15 hours[4]
3.8.9.D Two or more distribution subsystems (AC or DC) inoperable	1 hour[3]

Table E1-2 Notes:

1. The RICTs presented in this table are based on a Unit 1 model calculation. Due to the close similarity between the Unit 1 and Unit 2 models, the Unit 1 RICTs are considered adequate examples for the Unit 2 RICTs as well. Following 4b implementation, the actual RICT values will be calculated on a unit-specific basis, using the actual plant configuration and the current revision of the PRA model representing the as-built, as-operated condition of the plant, as required by NEI 06-09 and the NRC safety evaluation, and may differ from the RICTs presented.

2. RICTs are based on the internal events, internal flood, and internal fire PRA model calculations with seismic and extreme wind CDF and LERF penalties. RICTs calculated to be greater than 30 days are capped at 30 days based on NEI 06-09-A. RICTs are rounded to nearest number of days or hours for illustrative purposes.

3. This RICT is limited by the front-end Tech Spec completion time.

4. The limiting RICT for this Tech Spec was from the LERF calculation.

5. The saltwater cooling system provides bay water to the component cooling heat exchangers, the service water heat exchangers, and the Emergency Core Cooling System (ECCS) pump room air coolers. The service water system removes heat from turbine plant components, blowdown recovery heat exchangers, containment cooling units, SFPC heat exchangers, and Fairbanks Morse Emergency Diesel Generator heat exchangers. Therefore, complete TS inoperability of Service Water system does not have as significant an effect as complete TS inoperability of Salt Water system.

2. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238).

2. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

ENCLOSURE 2

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed
Completion Times TSTF-505, Revision 1, "Provide Risk-Informed
Extended Completion Times - RITSTF Initiative 4b."

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

1. Introduction

This enclosure provides information on the technical adequacy of the Calvert Cliffs Nuclear Power Plant (CCNPP) Probabilistic Risk Assessment (PRA) internal events model (including flooding) and the CCNPP fire PRA model in support of the license amendment request to revise Technical Specifications to implement NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" (Reference 1).

Topical Report NEI 06-09, Revision 0-A (Reference 1), as clarified by the NRC final safety evaluation of this report (Reference 2), defines the technical attributes of a PRA model and its associated Configuration Risk Management Program (CRMP) tool required to implement this risk-informed application. Meeting these requirements satisfies Regulatory Guide (RG) 1.174 (Reference 3) requirements for risk-informed plant-specific changes to a plant's licensing basis.

Exelon employs a multi-faceted approach to establishing and maintaining the technical adequacy and fidelity of PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the CCNPP PRA.

Section 2 of this enclosure describes requirements related to the scope of the CCNPP PRA internal events model. Section 3 outlines requirements for the internal events PRA from RG 1.200 (Reference 4) and how these are met. Section 4 similarly outlines requirements for the fire PRA from RG 1.200 and how these are met. Section 5 lists references used in the development of this enclosure.

2. Requirements Related to Scope of CCNPP Internal Events PRA Model

The CCNPP internal events PRA model is an at-power model (i.e., it directly addresses plant configurations during plant modes 1 and 2 of reactor operation). The model includes both at-power internal events core damage frequency (CDF) and large early release frequency (LERF). Internal flooding is included in both the CDF and LERF models.

Note that this portion of the CCNPP PRA model does not incorporate the risk impacts of external events. The treatment of seismic risk and other external hazards for this application are discussed in Enclosure 4.

3. Scope and Technical Adequacy of CCNPP Internal Events and Internal Flooding PRA Model

Topical Report NEI 06-09 requires that the PRA be reviewed to the guidance of RG 1.200 (Reference 4) for a PRA which meets Capability Category (CC) II for the supporting requirements of the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) internal events at power PRA standard (Reference 5). It also requires that deviations from these CCs relative to the Risk Informed Completion Time (RICT) Program be justified and documented.

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

The information provided in this section demonstrates that the CCNPP internal events PRA model (including flooding) meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 (Reference 4).

The PWROG performed a full scope internal events PRA and internal flooding PRA peer review of CCNPP to determine compliance with ASME PRA Standard, RA-S-2008, including the 2009 Addenda A (Reference 5) and RG 1.200 (Reference 4) in June 2010. This review documented findings for all supporting requirements (SRs) which failed to meet at least Category II. The findings for that peer review are documented in Table E2-1. This table also includes the status of disposition of those findings, and an assessment of impact on the CCNPP RICT program, if any.

The peer review found that 97% of the SR's evaluated Met Capability Category II or better. There were 3 SRs that were noted as "not met" and 8 that were noted as meeting Capability Category 1. As noted in the peer review report the majority of the findings were documentation related. Of the 11 SRs which did not meet Capability Category II or better, 7 were related to conservatisms or documentation in the LERF model and 2 were related to modeling of risk from internal floods. All findings have been dispositioned as described in Table E2-1 in the internal events model. With the exception of several documentation concerns, the internal events model meets Capability Category II for all SRs. As no new methods were applied in addressing the findings, no follow on or focused peer reviews were required.

Given the above, the CCNPP internal events PRA including internal flooding is of adequate technical capability to support the TSTF-505 program.

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

Table E2-1 CCNPP Internal Events PRA Peer Review – Facts and Observations - Findings

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to TSTF-505 Implementation
1-16	AS-B3 SY-B6	Systems Analysis	Based on Sections 2.4 and 2.10 of the System Analysis Introduction Notebook (C0-SY-00, Rev. 0) this SR appears to be met. However, there is a potential issue related to this SR. Did not find reference to any engineering analysis needed to support Containment Air Cooler operation when this system is assumed to be available during LOSP when the containment heats up prior to electrical recovery. (This F&O originated from SR SY-B6)	Complete	The PRA Internal Events Accident Sequence Notebook, C0-AS-001, Section 3.3, has been updated with an engineering analysis of this issue. The analysis identifies that during the Loss of Offsite Power sequences, the Containment Air Coolers are credited for SBO conditions where the containment heats up, and then, after power recovery, the air coolers are credited for containment pressure and temperature control. For these accident sequences, offsite power is restored in one hour, and the containment pressure and corresponding saturation temperature remain well below containment design parameters that would challenge the CACs. Furthermore, failure of CACs is not risk significant, due to the potential availability of containment spray.	This issue is resolved. It was determined that there was no impact on the FPIE PRA, and Subsequent internal events accident sequence analysis shows that Containment Air Cooler operation is not challenged by containment heat up during LOOP accident sequences that credit the CACs for recovery. Not an issue for RICT calculations.

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1-17	IFSO-A1 QU-E3	Internal Flooding	Examined Internal Flooding Notebook (C0-IF-001, Rev. 1) Sections 3.0 and 3.1. Part of the Internal Flood analysis may not be complete for assessing the Aux Feedwater Discharge Piping as a Flood Source. (This F&O originated from SR IFSO-A1)	Complete	An engineering analysis has subsequently been performed for AFW discharge piping flooding. The fraction of at-power time during which the AFW system is in operation 0.6% and the AFW Discharge Piping flood may be screened due to their low impact on CDF (<1E-9).	This issue is resolved. It was determined that there is no significant impact. Not an issue for RICT calculations.
1-18	IFSO-A4 IFEV-A7	Internal Flooding	Examined Internal Flooding Notebook (C0-IF-001, Rev. 1) Section 3.3 and 5.3. Consideration of human-induced mechanisms as potential flood sources not clear. Regarding human-induced impacts on the flood frequency, Section 5.3 of the IF report states that they were included, but their inclusion should be better documented or referenced from IF (e.g., a sample calculation showing human contribution would be helpful) (This F&O originated from SR IFSO-A4)	Complete	The Internal Flooding notebook now contains an extensive discussion of human-induced flood considerations.	This issue is resolved. Not an issue for RICT calculations.
1-19	IFEV-B3 IFPP-B3 IFQU-B3 IFSN-B3 IFSO-B3	Internal Flooding	While some items are included in Section 7.0 of the IF report, many other instances of uncertainties and assumptions are cited throughout the report, but not included in the discussion of Section 7.0 nor are the implications of these other uncertainties and assumptions are discussed. (This F&O originated from SR IFPP-B3)	Complete	Generic and model-specific uncertainties are included in the updated internal flood notebook and internal events uncertainty notebook.	This issue is resolved. Not an issue for RICT calculations.

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1-25	DA-C7	Data	For the most part actual plant-specific data is used as a basis for the number of demands associated actual plant experiences (See basis for DA-C6), which includes both actual planned and unplanned activities. However, there are a few ESFAS testing and/or other logic channel testing that are not tracked via the plant computer. Created this F&O on non-documentation of ESFAS/logic train testing, which needs to include actual practice. (This F&O originated from SR DA-C7)	Complete	The ESFAS logic train testing has a very low risk significance and generally does not take the logic OOS. Instead, the train goes to 2-out-of-3 logic. Occurrences where the train is in 2-out-of-3 logic are incorporated into the PRA Data Analysis Notebook, C0-DA-001, Section 2.6 and 3.5. For the logic relays there is a RAW of <1.04 and Birnbaum on the order of 4E-07. Any logic relay unavailability that does not cause the ESFAS channel to be OOS and bypassed is therefore of low significance.	The logic channel testing was determined to be low safety significant and the current Data analysis meets Capability Category II. Not an issue for RICT calculations.
2-7	IFPP-A5	Internal Flood	Section 2.3 provides a discussion that walkdowns used to confirm plant arrangement. The following note is contained in Section 2.3: Unfortunately, the walk-down documentation from the original flooding analysis no longer exists. A plant walk-down was performed as a part of this analysis to provide familiarity with the plant design as well as confirm findings from the original walk-down. This walk-down is documented in a set of notes and photographs included in Appendix B. Walkdown photos for room 105A and 203 show equipment and potential flood	Complete	A walkdown was performed to assess the susceptibility to jet impingement or spray in rooms 105A and 203. All equipment is considered failed by spray or impingement for flood sources originating in the room. Notebook C0-IF-001 was updated with this additional documentation.	This issue is resolved. Not an issue for RICT calculations.

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			propagation paths. However, there is not enough spatial information to develop specific targets for flood impingement or spray. (This F&O originated from SR IFPP-A5)			
2-9	DA-D4	Data	Evidence of meeting this SR at CC-II/III is found in the PRA Data Notebook (C0-DA-001, Rev. 1) in Sections 2.1 and 2.7. Found inconsistencies in the value of total number components of different types (for both units) in Table 2-5 of the PRA Data Notebook with the actual total number for Calvert Cliffs. Also, found an inconsistency between the prior distribution and posterior distribution for SACM EDG fail to start in Table 2-6 of the Data Notebook. (This F&O originated from SR DA-D4)	Complete	Table 2-6 of the Data Notebook listed incorrect data and Bayesian update results for the SACMs. However, the correct values were used in the models for peer review. For the SACM EDGs in Table 2-6, the correct plant-specific data are in Table 2-5. Table 2-6 lists incorrect data and Bayesian update results for the SACMs. However, the correct values are used in the models. The above errors have been corrected in the Data Notebook. Other minor typographical errors were identified and corrected in the notebook.	This was a documentation issue and has been resolved; the model includes the correct data. Not an issue for RICT calculations.
3-3	SY-C2	Systems Analysis	Section 2.3 of each system notebook states that marked up plant system drawings are provided as supplements to the system notebook, which depicts the boundary of the system in terms of PRA modeling. The drawings are not in the	Complete	Marked-up system boundary drawings were generated for each system notebook. Where Unit 1 and Unit 2 are similar, just the Unit 1 boundary is depicted. In addition, the	This was a documentation issue and has been resolved. Not an issue for

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			notebooks. (This F&O originated from SR SY-C2)		system notebooks include drawing snippets, sketches, and descriptive text that also depict the system boundary.	RICT calculations.
3-5	SY-A11 SY-A6	Systems Analysis	The fault tree does not include potential failures of the AFW accumulator system. (This F&O originated from SR SY-A11)	Complete	A bounding sensitivity case was run to include failure of the AFW accumulators failing short-term AFW operation. This issue has an insignificant contribution to CDF. Short-term failure of the AFW operation is dominated by failure of electrical support systems and failure of active hardware (i.e. valves and instrumentation). The applicable system notebooks were updated.	The random failure probability of the accumulators is two orders of magnitude lower than active hardware failures that support the same system function. This issue is resolved. Not an issue for RICT calculations.
3-8	SY-C1 SY-A13	Systems Analysis	Several system notebooks were reviewed (AFW, EDG, SI, 120 VAC electrical, etc.). In general, the documentation is complete and thorough. In most cases it clearly follows the RG 1.200 SRs. In some places, assumptions were imbedded in the documentation without sufficient reference or justification. Examples include: SI notebook page 11, last bullet 'Only one of the three HPSI pumps functions - For a cold leg break, it is assumed that only one-fourth pump discharge is spilled via	Complete	Some new flow diversions were identified as part of the Fire PRA Multiple Spurious Operation review, and these were added to the system models and system notebooks. Furthermore, a comprehensive review of PRA mechanical systems notebooks and drawings was performed to identify and document potential flow diversions. Flow diversion discussions were added to Sections 3.4.d of the	Flow diversion potential has been documented and addressed in the internal events PRA model. This issue is resolved. Not an issue for RICT calculations.

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			the break. For a hot leg break, the entire pump discharge reaches the core.' SI notebook page 12, 2nd bullet 'The maximum time assumed for operation for the safety injection pumps is 30 seconds following SIAS initiation.' C0-SY-000 states that each system notebook addresses flow diversions (where applicable) in Section 3.4.d. Although flow diversions appear to be addressed (for example, the SW notebook talks about flow diversion), there is no consistent discussion in each system notebook. (This F&O originated from SR SY-C1)		applicable system notebooks.	
3-9	DA-B1	Data	DA notebook table 2-5 contains the grouping of components for plant specific failure data. Many of the groupings appear to take into account differences in such things as size, type, mission type (e.g., FW TDP run vs. AFW TDP standby). However, in some cases, it is not clear what the basis for the grouping is. For example, SW MDP RUN and SRW MDP RUN are grouped together even though they are of different service conditions (salt water vs. clean water), voltages (480 VAC vs. 4160 VAC), size, etc. Similarly, AFW MDP is included with HPSI MDP and LPSI MDP, even though the two SI pumps are pumping borated water, while the AFW pump is pumping condensate grade water. No documentation of the	Complete	The model has been updated to add additional component types and failure modes to better reflect service conditions. Service Water and Salt Water pumps were broken out. AFW pumps and Safety Injection pumps were broken out. This resulted in changes to the associated failure rates. The change has been reflected in the Data Notebook.	This issue is resolved. Not an issue for RICT calculations.

			appropriateness of these groupings is provided. (This F&O originated from SR DA-B1)			
3-11	QU-B7	Quantification	The mutually exclusive cutsets for each system are described in the system notebook Section 3.4.e. Several SY notebooks were reviewed to determine appropriateness of the mutually exclusive cutsets. All appeared reasonable. A review was performed of the MUTEX gate within the fault tree model and the appropriate combinations identified in the SY notebooks appear have been included in the model. There are two gates under the MUTEX gate which contain mutually exclusive cutsets which are not documented in the system notebooks. While the majority of these are intuitively obvious (e.g., 11 Steam Generator Tube Rupture occurs as an IE AND 12 Steam Generator Tube Rupture occurs as an IE), these should be included in an appropriate system notebook. (This F&O originated from SR QU-B7)	Complete	A comprehensive review of mutual exclusive modeling was performed. Each system notebook and each system model was reviewed to validate the appropriateness of the modeling and reconcile any differences, and to verify that a documented basis exists for each mutually exclusive event. The PRA model was updated to reflect new, deleted, or re-organized mutually exclusive modeling identified as part of this review.	This issue is resolved. Not an issue for RICT calculations.
3-12	QU-D3	Quantification	A review of the top cutsets from each event tree was performed. The utility stated that during this review, cutsets were reviewed to determine if any mutually exclusive events were contained within cutsets, if any flag settings were inappropriate or if any recoveries were	Complete	Documentation of the cutset reviews was presented to the peer review team; although, the documentation was separate from the formal QU notebook package. A note was added to the QU notebook	This was a documentation issue and has been resolved. The original internal events cutset review notes have now been

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			overlooked or added inappropriately. A review of a sampling of cutsets did not indicate any inappropriate results. However, the QU notebook does not include a discussion of this review. (This F&O originated from SR QU-D3)		directing the reader to the location of the cutset review notes and spreadsheets. The PRA configuration control program requires a review of cutsets for PRA changes. In practice, the top CDF and LERF cutsets are examined for even the most innocuous model changes.	archived and internal events PRA cutset review sessions were conducted, documented, and archived. Not an issue for RICT calculations.
4-5	IE-A10 SY-A10 IE-C3 SC-A4	Initiating Events	The only mention in C0-SC-001 of shared systems between the units is the SBO EDG, noted in Section 4.1.2. It states that the SBO diesel can power any one bus on either unit. However, in the CAFTA model, there is an assumed bus preference of 11, then 24, then 12, then 23.* This is noted in the EDG system notebook but no basis is provided. The procedures do not actually have a preference, which yields a potentially non-conservative analysis. For example, if there is a LOOP, the U2 diesels fail to start and the U1 diesels fail to run after 1 hour. The SBO diesel would then be aligned to U2, and it is non-conservative to give the U1 bus 11 full credit. If such non-conservatism is negligible, some analysis should be performed to demonstrate this. (This F&O originated from SR IE-A10) *Note: Peer review finding was not precise. It should have stated bus	Complete	To address this finding, the Diesel Generator modeling was updated as described in the PRA DG System Notebook. EOP-7 directs to align the 0C DG to the unit with redundant safety equipment out-of-service, with a goal to restore at least one 4kV bus. Since 4kV Buses 11 and 24 support AFW, those busses would have a preference over Busses 14 and 21, all else being equal. No unit preference is modeled. If there is a conflict in the order-of-preference, for example, both 4kV Bus 11 and 4kV Bus 24 are not powered, then a 50-50 probability is assumed as to the preferred bus.	This issue is resolved. Not an issue for RICT calculations.

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			preference for Unit 1 is 11, then 24, and for Unit 2, is 24 then 21.			
4-12	HR-C1	Human Reliability	One basic event calculated in the appendix (ESF0HFCISZEFG) was not included in the fault tree models. CCNPP staff noted that it had previously been modeled, but inadvertently deleted in an update. (This F&O originated from SR HR-C1)	Complete	The basic event has been added to the model. A sensitivity run with the basic event included the current model showed no increase in risk. The system notebook C0-SY-048 was updated.	This issue is resolved. The missing basic event has been added to the internal events model. Not an issue for RICT calculations.
4-15	IFEV-A6	Internal Flooding	The internal flooding analysis did not have a formal process to gather plant specific design information, operating practices, etc. that could potentially affect the generic flooding frequencies. In response to an NRC RAI on the CCNPP ISI program plan, CCNPP mentioned a review of Condition Reports that did not find any items that would increase the flooding frequency. The CR review meets part of the requirement, but the SR also calls for reviews of plant design, operating practices, etc. that should be considered. The evaluation should be documented in the PRA. (This F&O originated from SR IFEV-A6)	Complete	An assessment of the site's design, operating practices, and other site-specific information was performed by a knowledgeable engineer. The review did not reveal evidence that flood likelihood at Calvert Cliffs should differ from generic industry data. A separate review of LERs turned up no instances of floods at Calvert Cliffs, further substantiating this observation. The design of Calvert Cliffs is not unique, the pumps used to pump bay water are located in the Intake Structure which is well isolated from the rest of the plant, and the plant is on the Chesapeake Bay but with a base elevation 45 feet above sea level, and the basement of	This issue has been resolved. Not an issue for RICT calculations.

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					the Turbine Building 12' above sea level. Water hammer events have occurred at Calvert Cliffs, but these events did not result in flooding. Based on the above, the generic flood rates are considered appropriate for Calvert Cliffs.	
4-19	LE-C13 LE-F3 LE-G4	Large Early Release	The sources of uncertainty are well identified in Table 5-1 of the LE notebook and quantified in Table 5-2 of the QU notebook. However, no discussion of the uncertainties or insights from them is provided. For example, Sensitivity 1 shows a 74% reduction in LERF, but this large reduction is not investigated. Also, conservatisms in the ISLOCA analyses were discussed in the AS review. SGTR was treated in an overly conservative manner by categorizing all SGTR as LERF. (This F&O originated from SR LE-F3)	Complete	Dominant LERF cutsets were reviewed to identify uncertainties that could be addressed. Two changes have been implemented to address significant uncertainties and reduced LERF. First, a reverse-flow check valve in the CVCS Letdown line was credited as a potential ISLOCA recovery. Second, a new human action was added with realistic timing for Steam Generator isolation and RCS depressurization on a SGTR. These and less significant model updates resulted in a LERF-to-CDF ratio change from approximately 17% to approximately 10%. This newer ratio is in the typical range for other PWRs.	This issue is resolved. Not an issue for RICT calculations.

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4-20	LE-F1 LE-G3	Large Early Release	The relative contribution to LERF is presented in the QU notebook by PDS and by initiating event, but not by accident progression sequence, phenomena, containment challenges or containment failure mode. (This F&O originated from SR LE-G3)	Complete	The contributions to LERF are documented in the Quantification Notebook and are noted as such in the Level 2 Notebook. The Level 2 notebook has been updated to point to additional phenomena and containment challenges and failure mode Tables/ Figures in the QU Notebook	This is a documentation issue and is resolved. Not an issue for RICT calculations.
4-21	LE-G5	Large Early Release	The LE notebook states that limitations in the LE analysis that could impact applications are documented in the QU notebook, but it is not. Given the conservative modeling of SGTR and ISLOCA, the impact on applications should include assessment of how this conservatism can skew the LERF results. (This F&O originated from SR LE-G5)	Complete	The Level 2 Notebook was revised to add a discussion of impact on results as part of the Unit 2 ILRT extension request.	This is a documentation issue and is resolved. Not an issue for RICT calculations.
4-22	LE-C10 LE-C12 LE-F2 LE-C3	Large Early Release	The LERF contributors have not been reviewed for reasonableness (per SR LE-F2). The QU notebook discusses the top 20 LERF cutsets (which total 73% of the total LERF). It notes conservatism in the cutsets and says it will be evaluated in Section 5.2, but is not. Section 4.3.6 of the QU notebook compares the total LERF of CCNPP to St. Lucie, but does not even break the results down by contributor (e.g., SGTR, ISLOCA, etc.). Also, the ASME PRA Standard SRs C-3,	Complete	The LERF results were reviewed for conservatisms as described in the SRs. After conservatisms were addressed (see discussion for F&0 4-19 above), no significant issues were identified.	This issue is resolved. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is similar to that for other PWRs. Not an issue for

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			C-10 and C-13 require a review of the LERF results for conservatism in the following areas: 1. Engineering analyses to support continued equipment operation or operator actions during severe accident progression that could reduce the LERF. 2. Engineering analyses to support continued equipment operation or operations after containment failure. 3. Potential credit for repair of equipment. No such review has been performed, despite the large conservatism noted in the containment bypasses. (This F&O originated from SR LE-F2)			RICT calculations.
5-10	LE-D7	Large Early Release	Following the failure of one or more containment penetrations to isolate on CIAS, a feasible operator action is to manually close the failed valves from the Main Control Room. (This F&O originated from SR LE-D7)	Complete	The merits of adding an operator action in order close containment penetration from the Main Control Room to recover from a containment isolation failure have been considered. A review of cutsets showed that a recovery is not feasible for top LERF sequences, because the sequence includes either 1) a loss of CR indication, 2) includes a station black-out condition, or 3) includes non-	Modeling of an operator action to manually close failed valves from the main control room would not significantly reduce LERF, as such an action is not feasible for the significant sequences where containment isolation has failed.

					recoverable pipe breaks.	This issue and is resolved. Not an issue for RICT calculations.
5-17	IE-C1 IE-C13 IE-C4	Initiating Events	Bayesian updates of non-time-based LOCA data were improper. The small and medium LOCA frequencies were obtained from draft NUREG 1829 then Bayesian updated (in App E) with CCNPP experience from 2004 to 2008. The Very Small LOCA prior having alpha = 0.4, Mean = 1.57E-03; was Bayesian updated to a Posterior having a mean value of 7.02E-04. This represents an excessive drop associated with CCNPP experience of 4 to 5 years. Similarly, the Small and Medium LOCAs were Bayesian updated with the whole industry experience rcy data. The draft NUREG 1829 LOCA frequencies were obtained from expert elicitations (not time-based) that included crack propagation analysis. The Bayesian update for VSLOCA used the Alpha parameter and the mean value to justify that the prior mean was based on 255 rcy. This may not have been the basis for the expert elicitations in NUREG 1829. Also, the Medium LOCA frequency may be classified as extremely rare event. It would require no Bayesian updating. The current CCNPP SLOCA and MLOCA frequencies are very close even though	Complete	The general concern on Bayesian updating of rare events is understood. However, the method used was based on a white paper developed by industry experts regarding LOCA frequencies. These experts included INL, NRC and Industry experts. In addition, the approach used for the Calvert PRA was the same as used for the NRC SPAR model. This issue is captured in the PRA configuration control database (CRMP).	The approach used for LOCA frequencies has been validated by industry experts and is the same approach as was used for the NRC's SPAR model. This issue and is resolved. Not an issue for RICT calculations.

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			the source data in NUREG 1829 indicates a negative exponential drop in these frequencies. (This F&O originated from SR IE-C1) (Note: rcy – reactor year)			
5-18	IE-C2 IE-C7	Initiating Events	Justify the exclusion of LOOP event at CCNPP in 1987. No time trend analysis was provided to justify the exclusion. (This F&O originated from SR IE-C2)	Complete	The event is not counted following guidance provided in NUREG/CR-6928, based upon trend analysis. A full discussion is included in the Initiating Event notebook.	This issue has been addressed. Not an issue for RICT calculations.
5-23	HR-A2	Human Reliability	The Pre-Initiator HRAs did not include the miscalibration of SIT pressure. For example, in the event where SIT pressure is miscalibrated high, various accident scenarios requiring SI are negatively impacted. Add SIT pressure miscalibrated high or, justify no impact on CDF / LERF. (This F&O originated from SR HR-A2)	Complete	It is agreed that the miscalibration of SIT pressure could have a negative impact on various accident scenarios involving LLOCA and VLLOCA initiators. However, this instrumentation is not modeled explicitly and is therefore deemed included within the component boundary for the SIT. As such the miscalibration probability would be included in the SIT unavailability.	The CCNPP SITs are only required and provide significant benefit on Large LOCAs. The frequency of a Large LOCA times the pre-initiator probability that would lead to SIT unavailability is negligible. This issue has been addressed. Not an issue for RICT calculations.
5-25	SC-C1 HR-I2 SC-C2	Success Criteria	Simplify the traceability of Tsw. In the post initiator HRA details, the HRA success criteria are often provided as a positive re-statement of the HRA title. And, the	Complete	Where applicable, the Tsw of each HFE that could be traced to the Success Criteria notebook was updated and	This is a documentation issue. This issue has been

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			consequence of failure is often stated as core damage. Consider adding Tsw to the success criteria and linking that to the PCTran case where Tsw was developed. Also, in the SC report (Table B-3), consider adding the actual time to core uncovery (or core damage) instead of providing a "Yes" entry in the column of "core damage?" (This F&O originated from SR HR-I2)		referenced in the HRA Calculator. The HRA notebook was also updated.	addressed. Not an issue for RICT calculations.
5-30	LE-D1 LE-B2	Large Early Release	Section 3.2.11 discussed the containment challenge from Hydrogen Combustion. It concluded that the challenge may be significant for some accident scenarios. The CCNP entry in Table 6.11-2 of the Level 2 WCAP showed a potentially significant impact from Hydrogen burn. Provide an estimate of the impact of Hydrogen burn on containment pressure. Use an accident scenario that is likely to produce larger amounts of H2 with failed containment spray. The optimal time to estimate the impact of Hydrogen burn is approximately at 2 hours which is the time when the EOF and TSC personnel have convened and are ready to guide the Main Control Room into periodic Hydrogen burns before the formation of explosive mixtures. (This F&O originated from SR LE-D1)	Complete	CCNPP's Level 2 PRA follows the analysis in WCAP-16341-P, Simplified Level 2 Modeling Guidelines. In the industry-supported analysis, the percentage of cladding oxidation is the main factor used to develop a maximum H_2 concentration in the containment, and, in turn, a containment pressure is calculated if the H_2 completely burns. These are then mapped to site-specific containment failure probabilities. A simplifying assumption is made that "no pre-burning of hydrogen generated in the core melt progression is considered." Calvert Cliffs' severe action management	The methodology in WCAP-16341-P is appropriate for Calvert Cliffs level 2 analysis for both internal events and fire initiators. This issue has been addressed. Not an issue for RICT calculations.

					procedures do include actions to reduce H_2 concentration in the containment, but these actions are not credited in the PRA model. Also, Containment Spray is not questioned for the LERF accident sequences. Containment Spray is a factor in LATE containment failure accident sequences.	
5-31	DA-D4	Data	The summary table for Bayesian updated parameters (on Page 53 of the PRA Data Notebook, C0-DA-001, Rev. 1) shows the CS-MDP was Bayesian updated with plant experience containing 1 failure and Zero run-hours. The CCNPP PRA staff responded to this issue as an isolated case. There is an actual FTR > 1 hr. (This F&O originated from SR DA-D4)	Complete	The aforementioned footnote was incorporated into Table 2-6 of the data notebook.	This is a documentation issue and no changes were required for the CS-MDP failure rate. This issue has been addressed. Not an issue for RICT calculations.
6-3	SC-B2	Success Criteria	Expert judgment was not used as the sole basis for any success criteria. However, upon inspection of the PCTran run tables in the SC report appendices, many instances of surrogate or inferred results were found. Instead of running specific PCTran calculations to cover the whole SLOCA break size spectrum, intermediate break sizes have been calculated supplemented with expert judgment to derive limiting time delay for operators to actuate SI (30 min) or limiting time delay	Complete	The approach for SLOCA break size analysis is discussed in the Success Criteria notebook. Furthermore, a review was conducted of this issue; in addition, TH analyses were completed to verify the break-size ranges. It was found that the computer simulations adequately represented the various break-size ranges.	The existing analysis meets the intent of the SR and therefore there is no impact on the PRA. This issue has been addressed. Not an issue for RICT calculations.

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			for OTCC (SGL<350'+10min). (This F&O originated from SR SC-B2)			
6-5	SY-A20	Systems Analysis	When appropriate, the simultaneous unavailability within a system is documented in the system notebooks and included in the PRA model. However, a further review of these items is required for completeness. (This F&O originated from SR SY-A20)	Complete	AFW basic event AFW0TMMAINT6-F7 was determined to not be needed in the plant model. The basic event was removed. All remaining AFW equipment unavailability events in the model and notebooks were reviewed for consistency. The review did not discover other missing or incorrect simultaneous unavailability events. AFW0TMMAINT-TF was determined to be modeled correctly, its description was found to be in error in the system notebook. The AFW System Notebook was updated. A review for concurrent maintenance was previously performed and documented in the Data Notebook.	This issue has been addressed. Not an issue for RICT calculations.
6-8	HR-H2	Human Reliability	Some recovery actions included in the model (thus credited) are set to screening values. In the HEP evaluation (appendices of the HR report) there are	Complete	For each screening HRA, the internal events analysis was updated to include a specific reference to the earlier HRA	The documentation for internal events HRAs was updated to address this

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			no indications that procedures, training, or other shaping factors are available on a plant-specific basis. (This F&O originated from SR HR-H2)		analysis. Included are the applicable success criteria for each recovery. Refer to Internal Events Human Reliability Analysis, and the associated HRA Calculator file.	issue. Internal events HRA with screening values were evaluated for applicability This issue has been addressed. Not an issue for RICT calculations.
6-9	HR-I1	Human Reliability	The HR report is well documented in general and will facilitate upgrades, however, some basic event names are not consistent between the HR report and the system notebooks. (This F&O originated from SR HR-I1)	Complete	Updated the notebooks in the reference section so HRA designator names and descriptions are the same in the HR Calculator, HR notebook, CAFTA Model 6.0. Changes included adding the "-B" extension and removing the "(-2)" event where applicable.	This is a documentation issue. HRA names in the model and notebook are now consistent. This issue has been addressed. Not an issue for RICT calculations.
6-10	IFPP-A2 IFSN-A2	Internal Flooding	Plant design features such as open rooms or as built divisions are used to define the flood areas and was well documented. More detail is needed as to why the containment buildings were screened from the analysis. (This F&O originated from SR IFPP-A2)	Complete	The Internal Flood notebook has been updated to incorporate an analysis describing the screening of the containment building from flooding analysis. Essentially, the containment is designed for LOCA condition, which screens reactor coolant system and related piping system. Other piping systems have	This is a documentation issue for the Internal Flood notebook. This issue has been addressed. Not an issue for RICT calculations.

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					limited inventory, are normally isolated, or have a low flow rate.	
6-14	IFSO-B1 IFSN-A9	Internal Flooding	While the flooding calculations have been performed and are thought to be correct and well done, additional documentation of data would enhance the IF report. It appears that the input reports and references are based on poorly documented or non-officially revisioned reports and information sources. (This F&O originated from SR IFSN-A9)	Open	This is a documentation finding for the internal floods notebook. The issue has been captured in the PRA configuration control database (CRMP). Using a graded approach based on the significance of the flood, some of the flood calculations have been re-evaluated with additional walk-downs and updated calculations. The configuration control item remains open.	This issue concerns the Internal Flood model and is a documentation issue. The documentation to resolve this F&O will be completed prior to the implementation of the RICT Program.
6-16	IFQU-A11 IFPP-B2	Internal Flooding	Walkdowns have been conducted and are documented in Appendix B of the IF report. It is stated in the IF report that prior information is no longer available; this fact should be corrected as required for analysis updates and information verifications. (This F&O originated from SR IFQU-A11)	Open	This is an internal floods documentation finding. The finding has been captured in the PRA configuration control database. Additional walkdowns have been performed for risk significant flood areas and scenarios. The configuration control item remains open.	This is a documentation issue concerning the Internal Flood model. The documentation to resolve this F&O will be completed prior to the implementation of the RICT Program.
6-17	IFQU-A10	Internal Flooding	By including the flooding events under the transient fault tree, the LERF impacts are automatically accounted for in the same manner as the general transient events in the LERF analysis. Very little	Complete	The impact of internal floods on LERF is included in the Quantification Notebook.	This issue has been resolved. Not an issue for RICT calculations.

			documentation is found related to the IF analysis in the LE report, although the IF report states that the LERF impacts due to flooding are documented and analyzed in the LE report. (This F&O originated from SR IFQU-A10)			
6-18	HR-H2	Human Reliability	The system time window Tsw for post initiator HRAs was frequently associated with 'core damage'. Post initiator HRAs that appear in the top cutsets may require success criteria linked to beginning of core uncovery (about 20 minutes before 'core damage'). Or, the operator actions that may fall into that final 20-minute time period should be overridden to assume a high stress level. While Section 3.1.5.7 described this approach, there is no evidence of its proper application in the HRA quantifications. (This F&O originated from SR HR-H2)	Complete	It was determined that the text in Section 3.1.5.7 was incorrect and does not capture how stress is actually applied in the EPRI HRA Calculator. The Internal Events PRA Human Reliability Analysis has been updated to show the stress level applied to each HFE and the justification for stress selection. Also included is a correlation between stress level and failure of execution probability.	The stress levels in the model are appropriate, but updates to the documentation are required. The internal events documentation was updated. This issue has been resolved. Not an issue for RICT calculations.
6-22	HR-E1	Human Reliability	Upon RAS, LPSI stops and EOP-5, Step S.1(d) requires the Operators to 'Shut RWT OUT Valves SI-4142, 4143'. This manual action was not modeled in the PRA. The CCNPP PRA staff provided reasonable response to this issue. Based on CR-2009-005581, there is no impact on pump operability. Also, the staff will continue to track the CR. If there are any changes to the disposition of pump operability, then a new HRA may be	Complete	As documented in CR-2009-005881, shutting the RWT outlet valves upon a RAS does not impact station operability. The Safety Injection Pumps and Containment Spray Pumps will not fail if the RWT isolation valves do not closed with a RAS signal. A design margin issue has been identified. This	The system is operable without the manual action to shut the RWT outlet valves. There is no impact on internal events. The issue was added to the plant's margin management program.

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			added to the PRA model (if warranted). (This F&O originated from SR HR-E1)		issue has been added to the plant's margin management program. No model changes have been made, but the PRA configuration management program would capture any design changes concerning this issue.	This issue has been resolved. Not an issue for RICT calculations.
6-23	HR-G7	Human Reliability	When the Calculator reads in the combinations, it assumes that actions occur in the order of the time delay (Td). However, the time delay is not the same for all sequences, and care must be taken to make the combinations appropriate for the sequences in which they occur. Page 88 of the HRA notebook indicates this was considered, since the Td was modified for events occurring prior to reactor trip, and also for OTCC after SG overfill. However, not all occurrences have been addressed. The combination examined by the review team is Combination 770 (OTCC after CST depletion). In this event the CST depletion should come first. (This F&O originated from SR HR-G7)	Complete	New HRA events, CVC0HFOTA8HRS-FR and AFW0HFCCSGDEC8HR-FR were added to model Td variances where CST depletion occurs early and when it occurs later. This account for appropriate sequencing of events.	This specific issue with time delay and CST depletion has been addressed and also incorporated into the internal events PRA model. This issue has been resolved. Not an issue for RICT calculations.
7-13	QU-A2	Quantification	Discrepancy between documentation and result files. SBO037 and SBO038 sequences appear to be inverted in Tables D-1, 4.2.2, 4.2.4, 4.2.5, B-3). (This F&O originated from SR QU-A2)	Complete	The top flood cutset was incorrectly flagged as being SBO sequence 37 (offsite power recovered < 1 hour) instead of sequence 38 (offsite power not recovered). Updated tables B-2, C-1, and D-1 in the	This is an internal events documentation issue. This issue has been resolved.

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		unit 1 quantification notebook. Spot-check was performed to identify other errors. In the unit 2 quantification notebook, fixed sequence 12 table 4.2-5, which incorrectly showed sequence 37 instead of 38.	Not an issue for RICT calculations.

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4. Scope and Technical Adequacy of CCNPP Fire PRA Model

The CCNPP Fire PRA (FPRA) peer review was performed January 2012 using the NEI 07-12 Fire PRA peer review process (Reference 7), the ASME PRA Standard, ASME/ANS RA-Sa-2009 (Reference 5) and Regulatory Guide 1.200, Rev. 2 (Reference 4). The purpose of this review was to establish the technical adequacy of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The 2012 Calvert FPRA peer review was a full-scope review of all of the technical elements of the CCNPP at-power FPRA against all technical elements in Part 4 of the ASME/ANS PRA Standard (Reference 5), including the referenced internal events supporting requirements (SRs). The peer review noted a number of facts and observations (F&Os). The finding F&Os and their dispositions are provided in Table E2-2. All findings have been dispositioned.

With the disposition of the peer review findings, the CCNPP FPRA meets at least Capability Category II for all applicable SRs. Eleven SRs were originally identified by the peer review team as meeting only Capability Category I requirements or as being "not met" for the requirement. An evaluation of the impact of those areas where only the Capability Category I requirement was met or the requirement was "not met" is provided in Table E2-3 along with the basis for now meeting Capability Category II.

Given the above, the CCNPP FPRA is of adequate technical capability to support the TSTF-505 program.

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
PP-B3-01	PP-B3 PP-B6 PP-C3	Plant Partitioning	Complete	The containment is partitioned into 2 PAUs. There are intervening combustibles and this was accounted for in the PRA by treating the 20 feet as an overlap region and failing components affected in both PAUs. There is no justification given for the 20 foot assumption. The turbine deck is continuous from unit 1 to unit 2. This area is divided into 2 PAUs, TURB1 and TURB2, but there is no discussion for the basis of the partitioning. Finding level of significance is based on crediting spatial separation with no requisite justification. Maintain the containment as 1 PAU and discern the separation of east from west in the fire modeling. Document the spatial separation and no intervening combustibles for the turbine deck.	C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook, was updated to include an analysis that justifies the partitioning of the containment into two plant partitioning units with a 20-foot spatial separation (known as the buffer zone). The only potential intervening combustibles in this buffer zone were identified as qualified cables that were verified to be encased within Marinite covered raceways. The covers prevent the cables from becoming potential combustibles and therefore are not considered intervening combustibles. The unit 1 and unit 2 Turbine Deck was walked down to assess for the acceptability of the Appendix R partitioning into distinct PAUs. The boundary was assessed to have at least a 20-foot separation between potential ignition sources and potential targets, assessed for intervening combustibles, and the Turbine deck volume assessed for damaging hot gas layer development. The partitioning was found acceptable and consistent with	This issue has been resolved. Not an issue for RICT calculations.

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
					NUREG/CR-6850, Section 1.5.2, where main turbine decks are typical applications where spatial separation has been credited.	
PP-B5-01	PP-B5 PP-C3	Plant Partitioning	Complete	The water curtain in the CCW room was credited as an active fire barrier. The justification was that the water curtain was part of the original regulatory fire protection program. This meets CAT 1, but needs enhancement for CAT II/III. Finding level was used because the requirements for CAT II/III were not met. Calvert Cliffs should provide a direct reference to their Appendix R program as the basis for the acceptability for this or provide a design basis justification for the water curtain and document that in the PP notebook if the Appendix R program reference cannot be found.	The Component Cooling Water room water curtain is an approved Appendix R exemption, as identified in the exemption issued by the NRC in response to Calvert Cliffs exemption request ER820816. The validity of crediting CCW Room Water Curtains is discussed in Southwest Research Institute Report No. 01-0763-201. A reference to the Southwest Research Institute report was added to C0-PP-001, Plant Partitioning Notebook.	This documentation issue has been resolved. Not an issue for RICT calculations.
PP-B7-01	PP-B7 PP-C3 PP-C4 QLS-A1	Plant Partitioning Qualitative Screening	Complete	1. The walk down nomenclature does not match the PP notebook. Example page 561 of the walkdown documentation	A table was created to correlate the building or area nomenclature that was used for the plant walkdown documentation, to the plant analysis	This issue has been resolved. Not an issue for RICT

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				uses nomenclature in the containment that does not match the PP notebook. 2. There are many areas inaccessible such as: #23 Charging Pump Room, U1 Service Water Pump Room, U1 East Battery Room, E/W Corridor. These areas appear to be accessible with a little effort. In some of the areas screened out in QLS, the areas were inaccessible and did not have a confirmatory walkdown. Finding level assessed due to the incompleteness of the walkdown documentation. 1. Prepare a table that correlates the PAUs from the PP notebook with the area nomenclature used in the walkdown documentation. 2. Complete the walkdowns, particularly for areas screened in the QLS task.	unit identifiers used in the Fire PRA analysis. This table was added to C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook. The facilities and rooms that were not originally walked-down were reviewed. Supplemental walkdowns were performed and supplemental walkdown datasheets were generated. For areas that were not accessible at the time of the supplemental walkdowns (for radiological safety reasons, personnel safety concerns, or access otherwise denied), The reason for inaccessibility was added to Table 17.	calculations.
CS-B1-01	CS-B1 CS-C4	Fire PRA Cable	Complete	Current Breaker coordination study still in progress. This	The breaker coordination study has been completed. PRA common	This issue has been resolved.

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
		Selection and Location		study needs to be completed in order to receive a Category II met for CS-B1. Complete the breaker coordination study.	power supplies are assumed to meet - the coordination requirements of NFPA 805, except as noted in C0-CS-001, Fire PRA Cable Selection Notebook.	Not an issue for RICT calculations.
PRM-B3-01	PRM-B3 PRM-B4 PRM-B5	Fire PRA/Plant Response Model	Complete	The FPRA model did not address events involving loss of both HVAC trains to the MCR, long term heatup of MCR and need for operator actions outside the MCR to compensate for the loss of electronic controls in the MCR, which was assumed as a CCDP of 1.0 for the plant. The basis for excluding this potential Core Damage sequence was addressed in questions to the Calvert Cliffs PRA team. This sequence is a new sequence outside the current model FPRA model logic trees. Consider using a combination of MCR heatup calculations to define the time when operators would leave MCR and consider a recovery action for restoring cooling the MCR.	Loss of Control Room HVAC can affect the operability and availability of equipment in the control room and cable spreading room. As described in Calvert PRA System Analysis Notebooks C0-SY-002, C0-SY-017, and C0-SY-030, loss of HVAC is modeled to have the effect of increasing the failure rate of 120VAC and 125VDC instruments and controls in the cable spreading room. For the control room, degradation of the 125VDC system is used as a conservative surrogate for control room I&C degradation. Loss of Control Room HVAC and subsequent temperature increases may adversely affect operator responses. The model reflects degradation of human actions by the degradation of the 125VDC system used for instruments and controls. Loss of Control Room HVAC is not	This issue has been resolved. Not an issue for RICT calculations.

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
					expected to cause abandonment by operations staff of the control room due to high temperatures. On complete loss of HVAC with no mitigation, such as no use of emergency fans, a calculation shows a CR temperature of 123 deg F at 24-hours. While this is a challenging environment, this temperature is assessed as insufficient to solely drive a complete CR abandonment scenario. NUREG/CR-6738 describes operational experience where operators will continue to occupy the control room even under severe environments. Operations staff says that in consideration of high temperatures in the control room, that Operations would do what was needed to keep the cores safe and covered. The site safety director says that for a temperature of 123 deg. F, the site would implement a mitigation strategy which would include stay-times, assessment of individuals for heat-related conditions, use of ice vests, and call-in of additional qualified operations staff to rotate into the control room.	

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
					The above discussion was included in C0-SY-030, Control Room HVAC PRA System Notebook.	
FSS-A5-01	FSS-A5	Fire Scenario Selection and Analysis	Complete	A range of ignition source / target set combinations has been represented for unscreened PAUs. These combinations are identified in relevant calculation sheets for unscreened PAUs. In some PAUs, sub-PAUs are defined and damage from a potential fire within the sub-PAU is addressed. However, it is not clear how or why damage would be limited to the specified sub-PAU because there are no physical barriers between specified sub-PAUs. The documentation is such that it cannot be determined if the selected fire scenarios provide reasonable assurance that the risk contribution of each unscreened PAU can be characterized. Another issue that influences the potential for fire propagation across sub-PAU boundaries is that the	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. In both cases, thermocouple location was adjusted as identified in F&O FSS-D3-02. For the CSR, consequences were divided into scenarios based on mitigation potential. First, if the scenario was suppressed by the Halon system then the limit of damage was based on what was predicted by FDS in terms of temperature and energy. If it was unsuppressed it went to total room burn, which assumes failure of all targets in the room, regardless of the initial scenario boundary. For the Switchgear Room FDS analysis, the analysis was updated to add clarity. A discussion of the application of sub-PAUs has been added to Addendum 1 to C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. Damage was not limited to specified sub-PAUs. Specific	This issue has been resolved. Not an issue for RICT calculations.

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				temperature measurement locations specified in the detailed FDS fire modeling evaluations do not generally coincide with locations where maximum temperature are expected (e.g., within the fire plume). As a consequence, for some fire scenarios damage to targets is not predicted when it should be based on the specified damage criteria. Some scenarios are screened on the basis of temperature measurements that do not represent conditions at targets within the fire plume. (See F&O FSS-D3-02) This could have a significant impact on the potential for fire propagation across sub-PAU boundaries and needs to be discussed more thoroughly.	examples of the treatment of fire growth and the application of sub-PAUs have been provided. As described in C0-FSS-004, the sub-PAU analysis included spatial information from walkdown, along with engineering judgment, to determine if fire sources could fail additional components, cables, or other combustibles, potentially leading to more damage to surrounding equipment or cables. For scenarios that leveraged FDT modeling, the issue related to whether the analysis had correctly addressed the impact of transients along the edge of a boundary interface for a sub-PAU. A comparable consideration was also related to secondary combustion and oil fires. Resolution involved selection of several representative PAUs for a sensitivity study that expanded the existing sub-PAUs and examined secondary ignition potential.	
FSS-A5-01	FSS-A5	Fire Scenario Selection	Complete	There were indications that Calvert Cliffs had the tools and information in place to properly	The PAUs were considered representative of the work performed based on several criteria. The	This issue has been resolved. Not an issue for RICT

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
		and Analysis		evaluate the propagation of fires across the sub-PAU boundaries given no physical barriers but there were no examples showing that this evaluation was performed or any explicit descriptions of how they were performed in general. The concern here is that without an explicit description of the process for evaluating the spread of fires across sub-PAU boundaries with no physical barriers and detailed examples, there is the potential that in the future, new people updating the PRA may not know that they have to evaluate this. Calvert Cliffs needs to describe their process for evaluating fire growth and propagation between sub-PAUs and as applicable, between PAUs. Specific examples of the sub-PAU fire growth need to be provided. If fire propagation from sub-PAU to sub-PAU was not treated, Calvert Cliffs needs to evaluate all sub-PAUs to determine if there is any potential for fire	analysis indicated that the methods mentioned were indeed appropriate. Sub-PAU impacts did not change from the expanded assessment and that secondary ignition was bounded by the existing analysis and was appropriately addressed. The analysis was incorporated into the documentation for C0-FSS-004.	calculations.

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				spread and then model the potential for spreading fires and for damage occurring across sub-PAU boundaries.		
FSS-D2-01	FSS-D2	Fire Scenario Selection and Analysis	Complete	Where used, the FDS model was generally used with a level of grid resolution that was below the level of grid resolution documented in the NUREG-1824 Verification and Validation study for the FDS model. A validation study was not conducted to support the use of this lower level of grid resolution. Grid resolution has a bearing on the results of FDS calculations. Grid resolutions outside the validation range in NUREG-1824 should be justified and validated. Increase the level of grid resolution in the FDS PAU Fire Evaluations (C0-FSS-004 R1) so that the grid resolution is within the validation range documented in NUREG-1824.	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. For the Cable Spreading Room FDS fire scenarios, a grid study was performed on the updated FDS model. The study recommended a grid size that was within the range in NUREG/CR-1824. That grid size was used for CSR FDS scenario evaluations. The study and results were incorporated into C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' Switchgear Rooms were updated to increase the level of grid resolution to a value that is within the validation range documented in NUREG/CR-1824. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as	This issue has been resolved. Not an issue for RICT calculations.

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
					Addendum 1.	
FSS-D3-01	FSS-D3 FSS-B2 FSS-D4	Fire Scenario Selection and Analysis	Complete	This SR is not met because detailed FDS fire modeling evaluations of PAUs 302, 306, 311, 317, 407 and 430 assume that material surfaces are "inert." As noted on p. 44 of C0-FSS-004 R1, this assumption was made "... so that no objects in the PAU or the PAU structure (walls, floor, or ceiling) itself would absorb any heat from the various fire scenarios, producing a more conservative or worst case result for all fire scenarios' impacts to the components and cables within the PAU model. As such, no detailed material properties were required to be defined in FDS for the scenarios to function correctly." However, specification of material surfaces as "inert" in FDS does not prevent heat absorption into material surfaces. On the contrary, this specification maintains material surfaces at ambient temperature in FDS, which tends to maximize heat	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. For the Cable Spreading Room FDS fire scenarios, the Unit 1 CSR was modified to include actual material properties and sensitivity analysis. Actual material properties were used in the updated U1CSR FDS model rather than the prior use of "inert" material conditions. Adiabatic conditions were used for any items with material properties that are unknown or of a high uncertainty to bound the analysis and prevent heat transfer into those objects. The CSR FDS model was executed and the results compared to the baseline results. This study was then documented in FSS-004. The results were applied to Unit 2 CSR. This study was then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' Switchgear Rooms were updated to specify	This issue has been resolved. Not an issue for RICT calculations.

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				absorption into these surfaces. To prevent heat absorption into material surfaces, they should have been specified as "adiabatic" rather than as "inert." The "inert" parameter in FDS maximizes heat transfer to surfaces rather than minimize it. This can result in lower calculated gas temperatures. Specify materials surfaces as "adiabatic" rather than as "inert" in FDS to prevent them from absorbing heat in order to achieve the stated goal of producing a more conservative or worst case result. This may prove to be overly conservative, in which case specification of realistic material properties could be used to achieve more realistic estimates of environmental conditions for these fire scenarios.	representative material properties as referenced by NUREG 1805. This adjustment enabled the analysis to obtain more realistic estimates of environmental conditions for these fire scenarios. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.	
FSS-D3-02	FSS-D3 FSS-A5	Fire Scenario Selection and Analysis	Complete	Temperature measurement locations specified in the detailed FDS fire modeling evaluations do not generally coincide with locations where	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms.	This issue has been resolved. Not an issue for RICT calculations.

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				maximum temperature are expected (e.g., within the fire plume). As a consequence, for some fire scenarios damage to targets is not predicted when it should be based on the specified damage criteria. Some scenarios are screened on the basis of temperature measurements that do not represent conditions at targets within the fire plume. Re-run FDS simulations with temperature measurement probes located within the fire plume or use other fire modeling tools such as FDTs to calculate fire plume temperatures for these scenarios.	For the Cable Spreading Room FDS fire scenarios, new measurement devices were included in the updated U1CSR FDS model. The thermocouples were placed directly above the fire source in the updated FDS model and the scenarios re-evaluated. The results were applied to Unit 2 CSR. This study and the results were then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' SWGR rooms were updated to alter the location of the thermocouples such that the centerline plume temperature was recorded and used to determine target impacts. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.	
FSS-D8-01	FSS-D8	Fire Scenario Selection and Analysis	Complete	Fire detection timing is evaluated for detailed fire modeling cases that use FDS. This fire detection timing is then used to estimate automatic fire suppression timing and fire brigade response timing for	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. For the updated Cable Spreading Room FDS fire scenarios, cable tray	This issue has been resolved. Not an issue for RICT calculations.

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				these scenarios. However, the fire detection timing is based on modeling that does not include obstructions located beneath the ceiling that could have an impact on fire detector response. The fire detection timing is also based on an unjustified assumption regarding the type of smoke detectors installed in the affected PAUs. Obstructions to the flow of fire gases can have an impact on smoke concentrations and velocities, which in turn influence smoke detector response. Without including such obstructions in fire modeling simulations, their impact on fire detection times is not evaluated. Include obstructions located beneath the ceiling for the affected fire scenarios in order to evaluate their impact on fire detection timing. Provide justification for the selection of the type of smoke detector specified in the FDS simulations for these fire scenarios.	obstructions were placed in the ceiling area of the updated U1CSR FDS model. Additional thermocouple and heat flux data recording devices were added to the U1CSR model under the new cable tray obstructions in the vicinity of the fire source. The scenarios were re-evaluated. The results were applied to Unit 2. A sensitivity study was also performed. The study and new scenario results were incorporated into C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' SWGR rooms were also updated to include significant obstructions such as cable trays and beam pockets within the switchgear rooms. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results and details of this analysis are documented in C0-FSS-004 as Addendum 1.	

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
FSS-F3-01	FSS-F3	Fire Scenario Selection and Analysis	Complete	To achieve CC II/III for this SR, a quantitative assessment of the risk of the selected fire scenarios involving a) exposed structural steel and b) the presence of a high-hazard fire sources must be completed consistent with the FQ requirements including the collapse of the exposed structural steel and any attendant damage. Such an assessment has not been done or was not documented in a readily discernible manner. This has a potential impact on fire risk quantification. Complete a quantitative assessment of the risk of the selected exposed structural steel fire scenarios consistent with the FQ requirements.	The Turbine Building was reviewed for potential fire scenarios where structural steel can be adversely affected. From the scenarios examined, those that can damage structural steel were selected for further analysis. The frequency, severity factor and non-suppression probability of each scenario were developed and included in the Structural Failure Analysis Notebook. These impacts were then added to FRANX database and quantified as part of the final Fire PRA risk quantification in Fire Quantification Notebooks C0-FRQ-001 and C0-FRQ-002.	This issue has been resolved. Not an issue for RICT calculations.
FSS-G4-01	FSS-G4	Fire Scenario Selection and Analysis	Complete	An assessment of the effectiveness, reliability and availability of credited passive fire barrier features has not been documented in the multi-compartment analysis. To achieve a CC II capability	Generic probabilities were used for credited passive fire barrier features in the multi-compartment analysis. At Calvert Cliffs, the fire barriers are verified to be effective through test procedures. An unreliability value was applied to all normally closed	This issue has been resolved. Not an issue for RICT calculations.

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				assessment, the effectiveness, reliability and availability of credited passive fire barrier features must be assessed. Assess the effectiveness, reliability and availability of credited passive fire barrier features and document this assessment.	doors that represents the probability of the door being propped open given a fire in the exposing compartment. The probability of finding a failed sealed wall penetration is assumed to be very small to warrant propagation scenarios. A discussion of the effectiveness, reliability, and availability of fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.	
FSS-G5-01	FSS-G5	Fire Scenario Selection and Analysis	Complete	The effectiveness, reliability and availability of credited active fire barrier features have not been quantified in the multi-compartment analysis. To achieve a CC II capability assessment, the effectiveness, reliability and availability of credited active fire barrier features must be quantified. Quantify the effectiveness, reliability and availability of credited active fire barrier features and document this assessment.	Active fire barriers were evaluated as effective in studies used to support Appendix R analysis. An unreliability value has been applied to all normally open, self closing dampers and doors; A discussion of the effectiveness of credited active fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.	This issue has been resolved. Not an issue for RICT calculations.
HRA-B2-01	HRA-B2	Human Reliability	Complete	Improve documentation of the adverse operator actions	C0-HRA-001, Fire Human Reliability notebook, was updated to detail the	This documentation issue has been

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
		Analysis		needed to address the impact of grounded or shorted electrical buses that might have an impact on other plant buses if not isolated and re-energized in the areas identified. Very difficult to find the information within the HRA notebook alone, because the actions are modeled as inputs to FRANX. Provide new tables listing the actions considered or references to specific locations.	adverse operator actions added to the model following the fire AOP review process. Table 3 was added to Section 2.2 detailing each basic event, set to true (1.0) used in the model to annotate the adverse operator actions in the model. These include actions to de-energize electrical busses to isolate them from potential shorts and grounds. Table 2 shows the HFEs added to the model as part of the AOP review, including actions to restore AC power to busses lost due to fire failure sequences.	resolved. Not an issue for RICT calculations.
HRA-E1-01	HRA-E1	Human Reliability Analysis	Complete	Documentation for what was done was very good, however, the details for not selecting any spurious alarms is not clear. The documentation of the adverse actions put into the model as "true" are not in the HRA report, actions identified in the cutset reviews are not clearly identified, rational for not using specific HFEs in the RCP trip actions, for identifying actions from procedures and the process for assigning uncertainty range for the	C0-HRA-001, Fire Human Reliability Notebook, was updated detailing the Alarm Response Procedure review process. Table 12 was expanded to show the ARP review of alarm impact and operator interview notes for CR annunciators that could result in a manual reactor trip. No annunciators were identified that would cause the operator to terminate a systems or components operation based solely on the alarm itself, but several were identified that could potentially result in the operator tripping the Unit	This documentation issue has been resolved. Not an issue for RICT calculations.

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				combos. Doesn't permit verification of the rational for judgments made in deciding what is in and out of the Fire HRA. Also, from the calculation viewpoint the need to know the use of all manpower requirements during early time after fire initiator for dependency analysis. Enhance documentation of the specific issues needed to reproduce the assumptions and calculations used in the HRA.	unnecessarily. C0-HRA-001 was also updated to detail the adverse operator actions added to the model following the fire AOP review process. Table 3 was added to Section 2.2 detailing each basic event, set to true (1.0) used in the model to annotate the adverse operator actions in the model. These include actions to de-energize electrical busses to isolate them from potential shorts and grounds. Table 2 shows the HFEs added to the model as part of the AOP review, including actions to restore AC power to busses lost due to fire failure sequences. New HFEs added as part of the cutset review process are identified in Table 1 of C0-HRA-001, Fire Human Action Reliability notebook. These are annotated with "identified during the development of the PRM Notebook." The cutset reviews are described in C0-QNS-001, Fire PRA Quantitative Screening Notebook. A new dependency analysis was performed after the new HFEs were added to the model, ensuring new	

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
					dependency combinations are considered. Additional information was added to Table 1 of the Human Reliability Analysis Notebook, C0-HRA-001, detailing why each HFE was either retained or removed. For example, event FGAFW0SGTRISOL, Operator Feeds Affected SG with SGTR to Assure Heat Removal, was "Not retained for fire scenarios, because these actions are SGTR specific. Modeling was not necessary to ensure these actions did not appear in the cut sets, because the SGTR initiator is not being used for fire scenarios." Combination event multipliers are used in cutsets of multiple HEP actions to account for dependencies between HEP actions. To account for the uncertainty in HEP actions, an uncertainty parameter is added to the HEP action. When performing uncertainty analysis, the uncertainty parameters for combination events is increased proportionally when they are multiplied by the combination event multipliers.	

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
					Based on interviews, there are sufficient non-control room personnel for fire recovery actions. Appendix D of C0-HRA-001 notes that there are no control room operators assigned to the fire brigade. There were no identified staffing issues or interferences between operators performing fire recovery actions and members of the fire brigade.	
FQ-A1-01	FQ-A1	Fire Risk Quantification	Complete	Treatment of 0 CCDPs scenarios is not clear and appears to result in an underestimate of total risk (the underestimate appears to be small based on the sensitivity evaluations performed): 1 - with respect to opposite unit quantification, use CCDP for reactor trip initiator unless confirmation of no trip is documented; 2 - address use of 0 CCDP for control room HVAC loss scenarios, apply CCDP consistent with control room abandonment	The fire risk quantification process has been updated in notebooks C0-FRQ-001 and C0-FRQ-002 to address the issue with FRANX fire scenarios having a zero conditional probability for CDF and LERF. 1. When documented analysis shows that selected fire scenarios for one unit are screened from impact for the opposite unit (typically, no trip would be initiated), then that scenario may be excluded from the opposite unit's fire risk quantification. Otherwise, a nominal conditional probability, as described in item 3 below, would apply. 2. F&O PRM-B3-01 identifies the	This issue has been resolved. Not an issue for RICT calculations.

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based on reactor trip initiator More than 50% of the scenarios have a 0 CCDP but no clear discussion of the basis for the 0 CCDP is provided. Treatment of 0 CCDPs scenarios: 1 - with respect to opposite unit quantification, use CCDP for reactor trip initiator unless confirmation of no trip is documented; 2 - address use of 0 CCDP for control room HVAC loss scenarios, apply CCDP consistent with control room abandonment 3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based	concern with loss of Control Room HVAC with control room abandonment. As discussed in more detail with the resolution to PRM-B3-01, subsequent investigation revealed that loss of CR HVAC is not expected to cause abandonment by the operations staff of the control room due to high temperatures. Loss of CR HVAC and subsequent temperature increases may adversely affect operator responses, and the model reflects degradation of human actions with loss of CR HVAC. C0-SY-030, Control Room HVAC PRA System Notebook, was updated to include this discussion. 3. The new quantification process described in the FRQ notebooks is to assure a nominal conditional value is calculated for these low significant scenarios by 1) recalculating the zero-conditional scenarios at a lower truncation value to assure resolution in the scenario cutset file and conditional probabilities , and/or to 2) use a baseline conditional probability for CDF and LERF for the internal events reactor trip initiating vent - IE0PT for Unit 1 or IE0PT-2 for	

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Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				on reactor trip initiator.	Unit 2	
FQ-B1-01	FQ-B1	Fire Risk Quantification	Complete	We observed zero CCDPs for some PAU CDF and LERF values in the FRANX tables (e.g., PAU 512) which eliminated loss of HVAC to the MCR as a potential MCR abandonment sequence. Treatment of 0 CCDPs scenarios: 1 - with respect to opposite unit quantification, use CCDP for reactor trip initiator unless confirmation of no trip is documented; 2 - address use of 0 CCDP for control room HVAC loss scenarios, apply CCDP consistent with control room abandonment (F&O FQ-A1-01 (F)) 3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based on reactor trip initiator	The fire risk quantification process has been updated in notebooks C0-FRQ-001 and C0-FRQ-002 to address the issue with FRANX fire scenarios having a zero conditional probability for CDF and LERF. 1. When documented analysis shows that selected fire scenarios for one unit are screened from impact for the opposite unit (typically, no trip would be initiated), then that scenario may be excluded from the opposite unit's fire risk quantification. Otherwise, a nominal conditional probability, as described in item 3 below, would apply. 2. F&O PRM-B3-01 identifies the concern with loss of Control Room HVAC with control room abandonment. As discussed in more detail with the resolution to PRM-B3-01, subsequent investigation revealed that loss of CR HVAC is not expected to cause abandonment by the operations staff of the control room due to high temperatures. Loss of CR HVAC and subsequent	This issue has been resolved. Not an issue for RICT calculations.

Table E2-2 CCNPP Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact to TSTF-505 Implementation
				Allowing zero CCDPs allows scenarios in the fire model to quantify with no contribution to the CDF or LERF value and this under represents those frequencies especially when considering delta risk evaluations. Replace the zero entries with the lowest CCPD for a plant trip with only random failures of the safety equipment as in the internal events model. We discussed this with the Calvert Cliffs PRA team and some of the zeros are due to fire areas in one unit potentially contributing to the CCDP of the opposite unit. With the exception of these cases a method for handling the zeros needed to be developed and applied in the frequency quantifications.	temperature increases may adversely affect operator responses, and the model reflects degradation of human actions with loss of CR HVAC. C0-SY-030, Control Room HVAC PRA System Notebook, was updated to include this discussion. 3. The new quantification process described in the FRQ notebooks is to assure a nominal conditional value is calculated for these low significant scenarios by 1) recalculating the zero-conditional scenarios at a lower truncation value to assure resolution in the scenario cutset file and conditional probabilities , and/or to 2) use a baseline conditional probability for CDF and LERF for the internal events reactor trip initiating vent - IE0PT for Unit 1 or IE0PT-2 for Unit 2	

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Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
PP-B3	2012 Peer Review: SR Not Met The containment is partitioned into 2 PAUs. There are intervening combustibles and this was accounted for in the PRA by treating the 20 feet as an overlap region and failing components affected into both PAUs. There is no justification given for the 20 assumption. The turbine deck is continuous from unit 1 to unit 2. This area is divided into 2 PAUs, TURB1 and TURB2, but there is no discussion for the basis of the partitioning. Associated F&O: PP-B3-01	Now: Met Cat II/III C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook, was updated to include an analysis that justifies the partitioning of the containment into two plant partitioning units with a 20-foot spatial separation (known as the buffer zone). The only potential intervening combustibles in this buffer zone were identified as qualified cables that were verified to be encased within marinate covered raceways. The covers prevent the cables from becoming potential combustibles and therefore are not considered intervening combustibles. The unit 1 and unit 2 Turbine Deck was walked down to assess for the acceptability of the Appendix R partitioning into distinct PAUs. The boundary was assessed to have at least a 20-foot separation between potential ignition sources and potential targets, assessed for intervening combustibles, and the Turbine deck volume assessed for damaging hot gas layer development. The partitioning was found acceptable and consistent with NUREG/CR-6850, Section 1.5.2, where main turbine decks are typical applications where spatial separation has been credited.	This issue has been resolved. Not an issue for RICT calculations.
PP-B5	2012 Peer Review: SR Met: (CC-I) The water curtain in the CCW room was credited as an active fire barrier. The justification was that the water curtain was part of the original regulatory fire protection program. This meets CAT 1, but needs enhancement for CAT II/III	Now: Met Cat II/III The Component Cooling Water room water curtain is an approved Appendix R exemption, as identified in the exemption issued by the NRC in response to Calvert Cliffs exemption request ER820816. The validity of crediting CCW Room Water Curtains is discussed in Southwest Research Institute Report No. 01-0763-201. A reference to the Southwest Research Institute report was added to C0-PP-001, Plant Partitioning Notebook.	This issue has been resolved. Not an issue for RICT calculations.

Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
	Associated F&O: PP-B5-01		
PP-B6	2012 Peer Review: SR Not Met The containment has a 20 foot area that overlaps between the E and W section. The overlap is specifically addressed in the PP notebook. The standard does not allow for an overlap. Associated F&O: PP-B3-01	Now: Met Cat I/II/III C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook, was updated to include an analysis that justifies the partitioning of the containment into two plant partitioning units with a 20-foot spatial separation (known as the buffer zone). The only potential intervening combustibles in this buffer zone were identified as qualified cables that were verified to be encased within marinate covered raceways. The covers prevent the cables from becoming potential combustibles and therefore are not considered intervening combustibles.	This issue has been resolved. Not an issue for RICT calculations.
CS-B1	2012 Peer Review: SR Met: (CC I) Supporting Requirement CS-B1 met with a category I. A breaker coordination study is currently being performed and is planned to be incorporated in the future. See Fact and Observation CS-B1-01. Associated F&O: CS-B1-01	Now: Met Cat II/III The breaker coordination study has been completed. As described in ECP-13-000321, Form 12, Engineering Evaluation, all PRA common power supplies are assumed to meet - or will meet - the coordination requirements of NFPA 805, except as noted in C0-CS-001, Fire PRA Cable Selection Notebook. As described in the cable selection notebook, two 120VAC lighting panels are not validated as coordinated, and these panels are assumed to fail for all Fire PRA scenarios. Also, as described in the PRA notebook a breaker for 480V motor control center MCC101BT has not been validated as coordinated. This breaker, 52-10150, is modeled so that a fire-induced electrical fault on the breaker's power cabling will fail MCC101BT. Finally, the notebook identifies that selected 120V power panels have coordination issues, but that these will be addressed by design changes and referenced in Attachment S – Modifications and Implementation Items.	This issue has been resolved. Not an issue for RICT calculations.

Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
PRM-B3	2012 Peer Review: SR Not Met No new initiating events were identified in the course of the fire PRA model generation. The failure of the control room HVAC does not lead to a control room abandonment CCDP (1.0 or other value justified by analysis as corresponding to shutdown from outside the control room) Associated F&O: PRM-B3-01	Now: Met Cat I/II/III Loss of Control Room HVAC can affect the operability and availability of equipment in the control room and cable spreading room. As described in Calvert PRA System Analysis Notebooks C0-SY-002, C0-SY-017, and C0-SY-030, loss of HVAC is modeled to have the effect of increasing the failure rate of 120VAC and 125VDC instruments and controls in the cable spreading room. For the control room, degradation of the 125VDC system is used as a conservative surrogate for control room I&C degradation. Loss of Control Room HVAC and subsequent temperature increases may adversely affect operator responses. The model reflects degradation of human actions by the degradation of the 125VDC system used for instruments and controls. Loss of Control Room HVAC is not expected to cause abandonment by operations staff of the control room due to high temperatures. On complete loss of HVAC with no mitigation, such as no use of emergency fans, calculation CA02725 shows a CR temperature of 123 deg F at 24-hours. While this is a challenging environment, this temperature is assessed as insufficient to solely drive a complete CR abandonment scenario. NUREG/CR-6738 describes operational experience where operators will continue to occupy the control room even under severe environments. Operations staff says that in consideration of high temperatures in the control room, that Operations would do what was needed to keep the	This issue has been resolved. Not an issue for RICT calculations.

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Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
		cores safe and covered. The site safety director says that for a temperature of 123 deg. F, the site would implement a mitigation strategy which would include stay-times, assessment of individuals for heat-related conditions, use of ice vests, and call-in of additional qualified operations staff to rotate into the control room. The above discussion was included in C0-SY-030, Control Room HVAC PRA System Notebook.	
PRM-B4	2012 Peer Review: SR Not Met See PRM-B3-01 F&O, not met due to no new initiators identified and the identification of a potential new initiator that was not quantified in the fire PRA model. Associated F&O: PRM-B3-01	Now: Met Cat I/II/III The potential new initiator has been assessed (failure of CR HVAC leading to CR abandonment as discussed in PRM-B3). Loss of Control Room HVAC and subsequent temperature increases may adversely affect operator responses. The model reflects degradation of human actions by the degradation of the 125VDC system used for instruments and controls. Loss of Control Room HVAC is not expected to cause abandonment by operations staff of the control room due to high temperatures. On complete loss of HVAC with no mitigation, such as no use of emergency fans, calculation CA02725 shows a CR temperature of 123 deg. F at 24-hours. While this is a challenging environment, this temperature is assessed as insufficient to solely drive a complete CR abandonment scenario. NUREG/CR-6738 describes operational experience where operators will continue to occupy the control room even under severe environments. Operations staff says that in consideration of high temperatures in the control room, that Operations would do what was needed to keep the cores safe and covered. The site safety director says that for a temperature of 123 deg. F, the site would implement a mitigation strategy which would include stay-times, assessment of individuals for	This issue has been resolved. Not an issue for RICT calculations.

Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
		heat-related conditions, use of ice vests, and call-in of additional qualified operations staff to rotate into the control room. The above discussion was included in C0-SY-030, Control Room HVAC PRA System Notebook.	
FSS-A5	2012 Peer Review: SR Not Met A range of ignition source / target set combinations has been represented for unscreened PAUs. These combinations are identified in relevant calculation sheets for unscreened PAUs (filenames RSC-CALKNX-2011-xxx.pdf). However, it is not clear how the potential for spreading fires and for fire and smoke spread between sub-PAUs is addressed and consequently it cannot be determined if the selected fire scenarios provide reasonable assurance that the risk contribution of each unscreened PAU can be characterized. Associated F&O: FSS-A5-01	Now: Met Cat I/II FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. In both cases, thermocouple location was adjusted as identified in F&O FSS-D3-02. For the CSR, consequences were divided into scenarios based on mitigation potential. First, if the scenario was suppressed by the Halon system then the limit of damage was based on what was predicted by FDS in terms of temperature and energy. If it was unsuppressed it went to total room burn, which assumes failure of all targets in the room, regardless of the initial scenario boundary. For the Switchgear Room FDS analysis, the analysis was updated to add clarity to the analysis. A discussion of the application of sub-PAUs has been added to Addendum 1 to C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. Damage was not limited to specified sub-PAUs. Specific examples of the treatment of fire growth and the application of sub-PAUs have been provided. As described in C0-FSS-004, the sub-PAU analysis included spatial information from walkdown, along with engineering judgment, to determine if fire sources could fail additional components, cables, or other combustibles, potentially leading to more damage to surrounding equipment or cables. For scenarios that leveraged FDT modeling, the issue related to whether the analysis had correctly addressed the impact of transients along the edge of a boundary interface for a sub-	This issue has been resolved. Not an issue for RICT calculations.

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Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
		PAU. A comparable consideration was also related to secondary combustion and oil fires. Resolution involved selection of several representative PAUs for a sensitivity study that expanded the existing sub-PAUs and examined secondary ignition potential. The PAUs were considered representative of the work performed based on several criteria. The analysis indicated that the methods mentioned were indeed appropriate. Sub-PAU impacts did not change from the expanded assessment and that secondary ignition was bounded by the existing analysis and was appropriately addressed. The analysis was incorporated into the documentation for C0-FSS-004.	
FSS-D3	2012 Peer Review: SR Not Met This SR is not met for multiple reasons. First, detailed FDS fire modeling evaluations of PAUs 302, 306, 311, 317, 407 and 430 assume that material surfaces are "inert." As noted on p. 44 of C0-FSS-004 R1, this assumption was made " so that no objects in the PAU or the PAU structure (walls, floor, or ceiling) itself would absorb any heat from the various fire scenarios, producing a more conservative or worst case result for all fire scenarios' impacts to the components and cables within the PAU model. As such, no detailed	Now: Met Cat. II. FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. <u>Material Properties</u> For the Cable Spreading Room FDS fire scenarios, the Unit 1 CSR was modified to include actual material properties and sensitivity analysis. Actual material properties were used in the updated U1CSR FDS model rather than the prior use of "inert" material conditions. Adiabatic conditions were used for any items with material properties that are unknown or of a high uncertainty to bound the analysis and prevent heat transfer into those objects. The CSR FDS model was executed and the results compared to the baseline results. This study was then documented in FSS-004. The results were applied to Unit 2 CSR. This study was then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook.	This issue has been resolved. Not an issue for RICT calculations.

Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
	material properties were required to be defined in FDS for the scenarios to function correctly." However, specification of material surfaces as "inert" in FDS does not prevent heat absorption into material surfaces. On the contrary, this specification maintains material surfaces at ambient temperature in FDS, which tends to maximize heat absorption into these surfaces. To meet the specified goal of preventing heat absorption into material surfaces, they should have been specified as "adiabatic" rather than as "inert." (See Finding FSS-D3-01) Second, temperature measurement locations specified in the detailed FDS fire modeling evaluations do not generally coincide with locations where maximum temperatures are expected (e.g., within the fire plume). As a consequence, for some fire scenarios damage to targets is not predicted when it should be based on the specified damage criteria. (See Finding FSS-D3-02)	The Unit 1 27' and 45' Switchgear Rooms were updated to specify representative material properties as referenced by NUREG 1805. This adjustment enabled the analysis to obtain more realistic estimates of environmental conditions for these fire scenarios. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1. <u>Temperature Measurement Locations</u> For the Cable Spreading Room FDS fire scenarios, new measurement devices were included in the updated U1CSR FDS model. The thermocouples were placed directly above the fire source in the updated FDS model and the scenarios re-evaluated. The results were applied to Unit 2 CSR. This study and the results were then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' SWGR rooms were updated to alter the location of the thermocouples such that the centerline plume temperature was recorded and used to determine target impacts. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.	

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Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
	Associated F&Os: FSS-D3-01 and FSS-D3-02		
FSS-F3	2012 Peer Review: SR Met: (CC I) A number of potential scenarios are selected and a qualitative assessment of the associated risk is performed for the selected fire scenarios. Associated F&O: FSS-F3-01	Now: Met Cat. II/III The subject of this SR is fire-induced damage to structural steel. As described in C0-FSS-005, Calvert Cliffs Fire PRA Structural Failure Analysis Notebook, the un-screened structural steel scenarios are in the Turbine Building. The Turbine Building analysis was reviewed for potential fire scenarios where structural steel can be adversely affected. From the scenarios examined, those that can damage structural steel were selected for further analysis. The frequency, severity factor and non-suppression probability of each scenario were developed and included in the Structural Failure Analysis Notebook. These impacts were then added to FRANX database and quantified as part of the final Fire PRA risk quantification in Fire Quantification Notebooks C0-FRQ-001 and C0-FRQ-002.	This issue has been resolved. Not an issue for RICT calculations.
FSS-G4	2012 Peer Review: SR Met: (CC I) Passive fire barriers are credited in the multi-compartment analysis consistent with fire resistance ratings, but the effectiveness, reliability and availability of credited passive fire barriers have not been assessed. Associated F&O: FSS-G4-01	Now: Met Cat. II Generic probabilities were used for credited passive fire barrier features in the multi-compartment analysis. At Calvert Cliffs, the fire barriers are verified to be effective through test procedures. An unreliability value was applied to all normally closed doors that represents the probability of the door being propped open given a fire in the exposing compartment. The probability of finding a failed sealed wall penetration is assumed to be very small to warrant propagation scenarios. A discussion of the effectiveness, reliability, and availability of fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.	This issue has been resolved. Not an issue for RICT calculations.

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Table E2-3 Fire PRA – Summary of Disposition of Capability Category I SR Assessments to Met or Category II

SR	Topic	Status	Impact to TSTF-505 Implementation
FSS-G5	2012 Peer Review: SR Met: (CC I) The effectiveness, reliability and availability of active fire barrier elements has been assessed qualitatively, but has not been quantified. Associated F&O: FSS-G5-01	Now: Met Cat. II/III Active fire barriers were evaluated as effective in studies used to support Appendix R analysis. An unreliability value has been applied to all normally open, self closing dampers and doors; A discussion of the effectiveness of credited active fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.	This issue has been resolved. Not an issue for RICT calculations.

5. References

1. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

2. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238).

3. Regulatory Guide (RG) 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011.

4. Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

5. ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RAS-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

6. NEI 05-04, "Process for Performing PRA Peer Reviews Using the ASME PRA Standard (Internal Events)," Revision 2, September 2008.

7. NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Revision 1, June 2010.

8. NUREG/CR-6850 (also EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," September 2005, with Supplement 1 (EPRI 1019259), September 2010.

ENCLOSURE 3

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

<u>Information Supporting Technical Adequacy of PRA Models Without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 2</u>

This enclosure is not applicable to the Calvert Cliffs Nuclear Power Plant submittal. Exelon is not proposing to use any PRA models in the CCNPP Risk-Informed Completion Time Program for which a PRA standard, endorsed by the NRC in RG 1.200, Revision 2 does not exist.

ENCLOSURE 4

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

1. Introduction and Scope

Topical Report NEI 06-09, Revision 0-A (Reference 1), as clarified by the Nuclear Regulatory Commission (NRC) final safety evaluation (Reference 2), requires that the License Amendment Request (LAR) provide a justification for exclusion of risk sources from the Probabilistic Risk Assessment (PRA) model based on their insignificance to the calculation of configuration risk as well as discuss conservative or bounding analyses applied to the configuration risk calculation. This enclosure addresses this requirement by discussing the overall generic methodology to identify and disposition such risk sources. This enclosure also provides the Calvert Cliffs Nuclear Power Plant (CCNPP) specific results of the application of the generic methodology and the disposition of impacts on the CCNPP Risk Informed Completion Time (RICT) Program. Section 3 of this enclosure presents the plant-specific bounding analysis of seismic risk to CCNPP. Section 4 of this enclosure presents the plant-specific bounding analysis of high wind risk to CCPP. Section 5 of this enclosure presents the justification for excluding analyses of other external hazards from the CCNPP PRA.

Topical Report NEI 06-09 does not provide a specific list of hazards to be considered in a RICT Program. However, NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Reference 4), provides regulatory guidance on risk-informed decision making relative to hazards that are not considered in the PRA model. Specifically, Section 6 of NUREG-1855 provides the following list of external hazards that should be addressed either via a bounding analysis or included in a PRA calculation:

Table E4-1
Minimum Scope of External Hazards to be Considered

- Seismic Events
- Accidental Aircraft Impacts
- External Flooding
- Extreme Winds and Tornados (including generated missiles)
- Turbine-Generated Missiles
- External Fires
- Accidents From Nearby Facilities
- Release of Chemicals Stored at the Site
- Transportation Accidents
- Pipeline Accidents (e.g., natural gas)

The scope of this enclosure is consideration of the above hazards for CCNPP. As explained in subsequent sections of this enclosure, risk contributions from seismic events and extreme winds and tornados are evaluated quantitatively, and the other listed external hazards are evaluated and screened as having low risk.

2. Technical Approach

The guidance contained in NEI 06-09 states that all hazards that contribute significantly to incremental risk of a configuration must be quantitatively addressed in the implementation of the RICT Program. The following approach focuses on the risk implications of specific external hazards in the determination of the risk management action time (RMAT) and RICT for the Technical Specification (TS) Limiting Conditions for Operation (LCOs) selected to be part of the RICT Program.

Consistent with NUREG-1855, external hazards may be addressed by:

1) Screening the hazard based on a low frequency of occurrence,
2) Bounding the potential impact and including it in the decision-making or
3) Developing a PRA model to be used in the RMAT/RICT calculation.

The overall process for addressing external hazards considers two aspects of the external hazard contribution to risk.

- The first is the contribution from the occurrence of beyond design basis conditions, e.g., winds greater than design, seismic events greater than design-basis earthquake (DBE), etc. These beyond design basis conditions challenge the capability of the SSCs to maintain functionality and support safe shutdown of the plant.

- The second aspect addressed are the challenges caused by external conditions that are within the design basis, but still require some plant response to assure safe shutdown, e.g., high winds or seismic events causing loss of offsite power, etc. While the plant design basis assures that the safety related equipment necessary to respond to these challenges are protected, the occurrence of these conditions nevertheless causes a demand on these systems that presents a risk.

Hazard Screening

The first step in the evaluation of an external hazard is screening based on an estimation of a bounding core damage frequency (CDF) for beyond design basis hazard conditions. An example of this type of screening is reliance on the NRC's 1975 Standard Review Plan (SRP) (Reference 5), which is acknowledged in the NRC's Individual Plant Examination of External Events (IPEEE) procedural guidance (Reference 6) as assuring a bounding CDF of less than 1E-6/yr for each hazard. The bounding CDF estimate is often characterized by the likelihood of the site being exposed to conditions that are beyond the design basis limits and an estimate of the bounding conditional core damage probability (CCDP) for those conditions.

If the bounding CDF for the hazard can be shown to be less than 1E-6/yr, then beyond design basis challenges from that hazard can be screened out and do not need to be addressed quantitatively in the RICT Program. The basis for this is as follows:

- The overall calculation of the RICT is limited to an incremental core damage probability (ICDP) of 1E-5.
- The maximum time interval allowed for this RICT is 30 days.
- If the maximum CDF contribution from a hazard is <1E-6/yr, then the maximum ICDP from the hazard is <1E-7 (1E-6/yr * 30 days/365 days/yr).
- Thus, the bounding ICDP contribution from the hazard is shown to be less than 1% of the permissible ICDP in the bounding time for the condition. Such a minimal contribution is not significant to the decision in computing a RICT.

The CCNPP IPEEE hazard screening analysis has been updated to reflect current CCNPP site conditions. The results are discussed in Section 5, and show that all the events listed in Table E4-1 can be screened except seismic events and extreme winds / tornadoes. Seismic and tornado hazards are addressed in Sections 3 and 4, respectively.

While the direct CDF contribution from beyond design basis hazard conditions can be shown to be non-significant using this approach, some external hazards can cause a plant challenge, even for hazard severities that are less than the design basis limit. These considerations are addressed in Section 5.

Hazard Analysis - CDF

There are two options in cases where the bounding CDF for the external hazard cannot be shown to be less than 1E-6/yr. The first option is to develop a PRA model that explicitly models the challenges created by the hazard and the role of the SSCs included in the RICT Program in mitigating those challenges. The second option for addressing an external hazard is to compute a bounding CDF contribution for the hazard. The approaches used for seismic and extreme wind risk are described in Sections 3 and 4, respectively.

Evaluate Bounding LERF Contribution

The RICT Program requires addressing both core damage and large early release risk. When a comprehensive PRA does not exist, the LERF considerations can be estimated based on the relevant parts of the internal events LERF analysis. This can be done by considering the nature of the challenges induced by the hazard and relating those to the challenges considered in the internal events PRA. This can be done in a realistic manner or a conservative manner. The goal is to provide a representative or bounding conditional large early release probability (CLERP) that aligns with the bounding CDF evaluation. The incremental large early release frequency (ILERF) is then computed as follows:

$$ILERF_{Hazard} = ICDF_{Hazard} * CLERP_{Hazard}$$

The approaches used for seismic and extreme wind LERF are described in Sections 3 and 4, respectively.

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Risks from Hazard Challenges

Given the selection of an estimated bounding CDF/LERF, the approach considered must assure that the RICT Program calculations reflect the change in CDF/LERF caused by the out of service equipment. For CCNPP, as discussed later in this enclosure, the only beyond design basis hazards that could not be screened out are the seismic hazard and the extreme winds hazard, and the approach used considers that the change in risk with equipment out of service will not be higher than the bounding seismic CDF or extreme winds CDF.

The above steps address the direct risks from damage to the facility from external hazards. While the direct CDF contribution from beyond design basis hazard conditions can be shown to be non-significant using these steps without a full PRA, there are risks that may be unaccounted for. These risks are related to the fact that some external hazards can cause a plant challenge even for hazard severities that are less than the design basis limit. For example, high winds, tornadoes, and seismic events can cause extended loss of offsite power conditions below design basis levels. Additionally, depending on the site, external floods can challenge the availability of normal plant heat removal mechanisms.

The approach taken in this step is to identify the plant challenges caused by the occurrence of the hazard within the design basis and evaluate whether the risks associated with these events are either already considered in the existing PRA model or they are not significant to risk.

Section 3 of this enclosure provides the analysis for the CCNPP site with respect to the beyond design basis seismic hazard, and Section 4 provides a similar analysis for the extreme winds hazard. Section 5 of this enclosure provides an analysis of the representative external hazards for the CCNPP site.

3. Seismic Bounding Analysis

This section presents the analysis that bounds the potential seismic impact for inclusion in the decision-making process, as a seismic PRA is not available for CCNPP. The process for analyzing an unscreened external hazard without the use of a full PRA involves the following three steps:

1. Estimate Bounding CDF
2. Evaluate Potential Risk Increases Due to Out of Service Equipment
3. Evaluate Bounding LERF Contribution

Estimate Bounding CDF

A seismic PRA (SPRA) was developed for the CCNPP Individual Plant Examination for External Events (IPEEE), Reference 7, however, that model has not been maintained as a current SPRA. Therefore, an alternative approach is taken to provide an estimate of seismic core damage frequency (SCDF) based on the current CCNPP seismic hazard curve and assuming the seismic capacity of a component whose seismic failure would lead directly to core damage. This approach to estimation of the SCDF uses a plant level high confidence of low probability of failure (HCLPF) seismic capacity and convolves the corresponding failure probabilities as a function of seismic hazard level with the seismic hazard curve. This is a commonly used approach to estimate SCDF when a seismic PRA is not available. This approach is consistent with approaches that have been used in other regulatory applications.

The seismic hazard for the CCNPP site was evaluated in 2013 (Reference 8) and provided to NRC via Reference 9. The CCNPP IPEEE identified a plant level HCLPF of 0.3g PGA (peak ground acceleration) consistent with the CCNPP IPEEE review level earthquake (RLE). Subsequently a limiting HCLPF value of 0.27g PGA was defined in calculations performed in support of Reference 28, and that lower HCLPF is selected for use in determining the bounding SCDF. Calculation of the SCDF in this manner also requires definition of uncertainty parameters for seismic capacity. The uncertainty parameter for seismic capacity is represented by a combined beta factor (βc) of 0.4. This is a commonly-accepted approximation, and is consistent with the value used in Reference 10. Using the above inputs, the total estimated CCNPP SCDF is determined to be 8.4E-7 for the IPEEE (0.3g PGA HCLPF) case and 1.1E-6/yr for the EDG AOT submittal (0.27g HCLPF) case. The higher SCDF value will be used as the bounding estimate of SCDF ($ICDF_{seismic}$) for the TSTF-505 submittal RICT calculations.

Evaluate Potential Risk Increases Due to Out of Service Equipment

The approach taken in the computation of SCDF assumes that the SCDF can be based on the likelihood that a single seismic-induced failure leads to core damage. This approach is bounding and implicitly relies on the assumption that seismic-induced failures of equipment show a high degree of correlation (i.e., if one SSC fails, all similar SSCs will also fail). This assumption is conservative, but direct use of this assumption in evaluating the risk increase from out of service equipment could lead to an underestimation of the change in risk. However, if one were to assume no correlation at all in the seismic failures, then the seismic risk would be lower than the risk predicted by a fully correlated model, but the change in risk using the un-correlated model with a redundant piece of important equipment out of service would be equivalent to the level predicted by the correlated model.

If the industry accepted approach (Reference 11) of correlation is assumed, the conditional core damage frequency given a seismic event will remain unaltered whether equipment is out of service or not. Thus, the risk increase due to out of service equipment cannot be greater than the total SCDF estimated by the bounding method used in Reference 11. That is, for the CCNPP site, the delta SCDF from equipment out of service cannot be greater than 1.1E-6/yr.

Evaluate Bounding Seismic LERF Contribution

The current CCNPP internal events PRA (Reference 12) includes a comprehensive treatment of LERF due to internally initiated events. The internal events PRA provides an estimate of the conditional probability of LERF for each modeled initiating event. Seismic events would not be expected to induce containment bypass scenarios, e.g., Steam Generator Tube Rupture (SGTR) or Interfacing Systems Loss of Coolant Accident (ISLOCA), and the bypass resulting from SGTR or ISLOCA is not a function of containment seismic capability. Therefore, a bounding conditional large early release probability for seismic events (CLERPSeismic) can be obtained by examining the event-specific CDF and event-specific LERF, for the non-direct bypass events, i.e.,

$$CLERP_{IE} = LERF_{IE} / CDF_{IE}$$

Using the current CCNPP internal events PRA, the average CLERP over all initiating events other than direct containment bypass events is approximately 6%. A 10% value of CLERP encompasses those internal events initiators contributing over 95% of total LERF and total CDF, and is selected for use in determining representative RICTs for this LAR as an adequately conservative but not overly pessimistic estimate. Therefore, CLERPSeismic = 0.1.

The incremental bounding large early release frequency from seismic events (i.e., the SLERF) is then computed as:

ILERFSeismic = ICDFSeismic * CLERPSeismic = 1.1E-6 * 0.1 = 1.1E-7

Since this estimation of CLERP may change as the internal events PRA model is updated, the estimate will be updated for the RICT program with each internal events model update.

Conclusion

The above analysis provides the technical basis for addressing the seismic-induced core damage risk for CCNPP by reducing the ICDP/ILERP criteria to account for a bounding estimate of the configuration risks due to seismic events.

The RICT and RMAT calculations are based the discussion provided above. The actual RICT and RMAT calculations performed by the CCNPP Configuration Risk Management Tool are based on adding an incremental 1.1E-6/year seismic CDF contribution and corresponding seismic LERF contribution to the configuration-specific delta CDF and delta LERF attributed to internal and fire events contributions. This is accomplished by adding these seismic contributions to the instantaneous CDF/LERF whenever a RICT is in effect. This method ensures that an incremental seismic CDF/LERF equal to the bounding SCDF/SLERF is added to internal and fire events incremental CDF/LERF contribution for every RICT occurrence.

4. Extreme Winds Analysis

This section presents the analysis that conservatively accounts for the risk associated with high winds and wind-driven missiles. A conservative approach is used since a peer reviewed PRA is not available for this hazard for CCNPP. The analysis focuses on the tornado hazard to CCNPP, primarily the effect of tornado-induced missiles. Hurricanes and straight winds are screened from consideration in the RICT Program based on the following considerations:

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- The hurricane hazard to CCNPP is screened due to the procedures in effect that direct a plant shutdown to Mode 3 when a hurricane is expected to arrive within 8 hours (Reference 13). Since both CCNPP units will be in Mode 3 at the time of a hurricane strike on the site, the RICT Program is not applicable. Therefore, the hazard need not be accounted for in this application.

- The straight wind hazard includes high winds primarily from thunderstorms and extratropical storms. Due to the lower wind speeds involved in these events, the primary concern is loss of offsite power (LOOP). There is industry experience with winds causing siding (such as from a turbine building) to become missiles during high wind events. However, these are lightweight missiles that generally do not damage safety related SSCs or other engineered structures (e.g., tanks); the primary concern is a LOOP. Since the internal events PRA includes LOOP events due to severe weather, the hazard associated with straight winds is considered in the RICT calculations and need not be accounted for separately.

Tornado winds and missiles are evaluated using a version of the internal events PRA that considers the likelihood of tornado wind and missile failures of susceptible SSCs. The tornado missile analysis is consistent with the analysis used in the CCNPP UFSAR, Section 5A (Reference 14). In the UFSAR, unprotected SSCs are evaluated for their contribution to CDF using a simplified and conservative tornado missile analysis, previously approved by NRC for use at CCNPP (Reference 15). In this approach, certain unprotected SSCs are considered to fail (due to winds and/or missiles) with a probability of 1.0 during a tornado event. For the remaining unprotected SSCs, the conditional missile strike probability, P_{ms}, is calculated by:

$$P_{ms} = A\ N_m\ \psi$$

Where A is the target area,
N_m is the number of missiles on the site, and
ψ is the missile impact parameter (probability of impact/missile/sq-ft target area/tornado strike frequency)

A conservative assumption is made that if a target is struck by a tornado missile, it is considered to fail with a probability of 1.0 (except for the main steam safety valve, MSSV, and atmospheric dump valve, ADV, vent stacks). The areas for the various SSCs and their conditional missile strike probabilities are used with the internal events PRA model, based on the values provided in Reference 16.

The tornado strike frequency (the likelihood that a tornado hits the site) is based on the NUREG/CR-4461 (Reference 17) tornado strike frequency for the 2° x 2° box containing CCNPP.

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Estimate Conservative Extreme Wind CDF and LERF

Using the CCNPP internal events PRA model, incorporating conservative assumptions and conditional missile strike probabilities, the CDF and LERF for both Units 1 and 2 are estimated. Table E4-2 provides the results for the average maintenance and zero maintenance (0M) base cases.

Table E4-2
CCNPP Tornado Risk

	Average CDF	Average LERF	0M CDF	0M LERF
Unit 1	3.3E-7/yr	1.6E-8/yr	2.5E-7/yr	1.2E-8/yr
Unit 2	5.4E-7/yr	2.9E-8/yr	4.7E-7/yr	2.6E-8/yr

The base tornado CDF values for both units are below 10^{-6}/yr. However, tornadoes are not screened on this basis, since the unavailability of SSCs may result in significant incremental risk increases.

Evaluate Potential Risk Increases Due to Out of Service Equipment

The set of TS under consideration for RICT evaluation (refer to Enclosure 1) were each evaluated individually by assuming the equipment associated with the TS is unavailable. The analysis described above was used, considering all SSCs were available (i.e., zero maintenance) with the exception of the TS equipment. The increases in CDF and LERF for both units were estimated.

The most limiting of the tornado CDF and LERF increases (with the exception of the eleven TS LCOs listed below), are used to define bounding values for application to all TS RICT evaluations. This is conservative, in that many SSCs affected by TS have only minor impacts on tornado risk, if unavailable (Reference 18).

The eleven LCOs excluded from this assessment are listed below:

- 3.3.4.B Two ESFAS Modules or Measurement Channels Inoperable
- 3.3.5.E Two ESFAS Actuation Logic Trains Inoperable
- 3.7.4.A CST inoperable (CST 12 only)
- 3.7.6.C Two SRW subsystems inoperable
- 3.7.7.B Two SW subsystems inoperable
- 3.8.1.J Three or more offsite and DGs inoperable (3 or more DGs only)
- 3.8.4.A One Battery Inoperable and Reserve Battery Available
- 3.8.4.B One DC Channel Inoperable
- 3.8.4.C Four DC channels inoperable
- 3.8.9.C One DC Distribution Subsystem Inoperable (Bus 11/21 only)
- 3.8.9.D Two or more distribution subsystems (AC or DC) inoperable

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Four of these LCOs (3.3.4.B, 3.7.4.A, 3.8.4.C and 3.8.9.D) were excluded because the RICTs calculated using the internal events PRA and fire PRA models resulted in RICTs less than or equal to the front stop CTs, so it would be overly pessimistic to assign a bounding tornado risk penalty based on these four configurations to RICT calculations for all other TS conditions covered by this program.

For the other seven LCOs, the configuration-specific delta CDF/LERF from extreme winds is much less than the delta CDF/LERF from internal events, fire, and seismic, sometimes by a factor of roughly 100. However, the extreme wind delta CDF/LERF in these cases would be too conservative to use as a penalty on all RICTs. For example, for 3.8.4.B, the limiting non-wind ICDF = 1.8E-3 and the extreme wind ICDF = 1.2E-5, so the wind impact on the RICT is not significant in this case and it would not be appropriate to assign this configuration-specific risk as a penalty on all RICTs.

Furthermore, the CDF/LERF calculations done for some of the extreme wind cases assume complete loss of PRA functionality (e.g., 3.7.6.C, 3.7.7.B, and 3.8.4.C).

Table E4-3 provides the maximum CDF and LERF increases for Units 1 and 2, with the exception of the eleven TS LCO listed above.

Table E4-3
Maximum Risk Increase for Tornado Hazard

	CDF Increase	LERF Increase
Unit 1	3.0E-6/yr	3.7E-7/yr
Unit 2	4.3E-6/yr	4.5E-7/yr

One of the significant differences between Unit 1 and 2 with respect to tornado missile risk is the dependency of both Unit 2 DGs on service water (SRW), for which there is a tornado missile impact. In Unit 1, the 1A DG is not dependent on SRW, so the types of tornado missile impacts that might fail both Unit 2 DGs would only cause a loss of one Unit 1 DG. In order to provide consistency between the two risk metrics and the two units (and providing an additional measure of conservatism), the increase in risk associated with tornadoes to be accounted for in the RICT evaluation for all but the eleven listed TS will be conservatively rounded up from the values in Table E4-3 to be 5E-6/yr for CDF and 5E-7/yr for LERF. The implementation of adding this additional risk in the RICT evaluation uses the same process described for seismic risk in Section 3. That is, the RICT and RMAT calculations performed by PARAGON (the CCNPP Configuration Risk Management Tool) are based on adding an incremental 5E-6/year tornado CDF contribution and 5E-7/year tornado LERF contribution to the configuration-specific delta CDF and delta LERF attributed to internal and fire events contributions (and the seismic ICDF/ILERF). This is accomplished by adding these tornado contributions to the PARAGON logic model that is used to quantify instantaneous CDF/LERF whenever a RICT is in effect. This method ensures that an incremental tornado CDF/LERF equal to the bounding values is added to internal and fire events incremental CDF/LERF contribution for every RICT occurrence.

Since the penalty factor does not bound the extreme wind risk for the eleven LCOs discussed above, additional analyses or restrictions will be required during RICT implementation, in the unlikely event a RICT evaluation is performed for any of those LCOs. For example, entering an extended completion time may be precluded during time periods with severe weather forecasts, or a bounding penalty factor may be applied.

Since these tornado CDF and LERF values were determined using the current internal events PRA model, they will need to be re-evaluated when changes are made in the internal events PRA model (which serves as the basis for the estimated tornado risk) or if changes occur to the tornado hazard (e.g., an update to NUREG/CR-4461, Reference 17).

5. Evaluation of External Event Challenges and IPEEE Update Results

The primary purpose of this section is to address the incremental risk associated with challenges to the facility that do not exceed the design capacity. This section also provides results of the hazard screening described earlier. Seismic events and extreme wind events are the only hazards not screened out.

In accordance with NUREG-1855 (Reference 4), Table E4-1, above, lists the external hazards considered.

Hazard Screening Except Seismic and Extreme Wind Events

The CCNPP IPEEE for Units 1 and 2 (Reference 7) provides an assessment of the risk to CCNPP associated with these hazards. Additional analyses have been done since the IPEEE to provide updated risk assessments of various hazards, such as aircraft impacts (Reference 19), industrial facilities and pipelines (Reference 20), and external flooding (Reference 21).

External hazards other than seismic and high winds can be screened for the CCNPP site. The basis for screening the other external hazards is presented in Table E4-4.

Risks from Hazard Challenges Other Than Seismic and Extreme Wind Events

Table E4-4 reviews the bases for the evaluation of these hazards, identifies any challenges posed, and identifies any additional treatment of these challenges, if required. The conclusions of the assessment, as documented in Table E4-4, assures that the hazard either does not present a design-basis challenge to CCNPP, or is adequately addressed in the PRA.

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Seismic-Induced Loss of Offsite Power Challenges

For the CCNPP site, the only incremental risk associated with challenges to the facility that do not exceed the design capacity, which is not already addressed, is the seismically-induced LOOP. The methodology for computing the seismically-induced LOOP frequency is simply a convolution of the mean seismic hazard curve and the offsite power fragility. The CCNPP seismic hazard curve is as described in Section 3.

Table E4-5 provides the mean seismic hazard, represented by a series of discrete seismic hazard intervals from just below the CCNPP operating basis earthquake to significantly above the safe shutdown earthquake, and the LOOP failure probability for each seismic interval based on the fragility of offsite power, represented by failure of ceramic insulators in the offsite power switchyard. The failure probabilities are based on the fragility data from Table 4B-1 of the RASP Handbook (Reference 27):

Median Offsite Power Capacity = 0.3g, β_R = 0.3, β_U = 0.45

Given the mean frequency and failure probability for each seismic hazard interval, the estimated frequency of seismically induced loss of offsite power for the CCNPP site is obtained by taking the product of the interval frequency and the offsite power failure probability. As shown in Table E4-5, the total seismic LOOP frequency is the sum of interval frequencies, or 1.2E-5/yr.

The internal events PRA models LOOP from plant-centered, switchyard-centered, grid-related, and weather-related events. Based on the CCNPP internal events PRA, total frequency of unrecovered loss of offsite power, i.e., the sum of the frequency times the non-recovery probability at 24 hours over these LOOP events, is 1.8E-3/yr.

The seismically-induced (unrecoverable) LOOP frequency is therefore less than 1% of the total unrecovered LOOP frequency that is already accounted for in the internal events PRA. This frequency is judged to be a sufficiently small fraction that it will not significantly impact the RICT Program calculations and it can be omitted.

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Table E4-4
Evaluation of Risks from External Hazards

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
Seismic Events	Seismic events treated using a bounding approach with change to RICT Program criteria (see Section 3 of this enclosure).	Seismically induced loss of offsite power (LOOP) is a challenge within the design basis.	Addressed as part of internal events treatment of LOOP.
Accidental Aircraft Impacts	There are several airways and small airports within 10 miles of the site. A military airfield is located just beyond the 10-mile radius. A post-IPEEE analysis of aircraft crash rates for CCNPP was performed in 2002; the analysis determined that the aircraft crash rate into safety related structures is less than 10^{-6}/yr (Reference 19). Note that this includes all aircraft (including general aviation aircraft that are typically lightweight); therefore, the core damage probability for aircraft crashes is less than 1.0. A review of recent trends in air traffic (increase) and aircraft crash rates (decrease) was performed to verify that the analysis in Reference 19 continued to be applicable. One of the primary contributors to the aircraft crash rate calculations at CCNPP is local helicopter operations at the site helipad. Therefore, the crash rate calculated in Reference 19 is contingent on	Aircraft impact induced LOOP is a potential challenge within the design basis. Additionally, aircraft crashes in other areas of the site (i.e., non-safety related areas or structures) may lead to fires.	The risk from aircraft crashes into safety related structures can be excluded from RICT Program evaluation due to the low frequency of these events. The likelihood of a LOOP or plant fire due to aircraft crashes in other areas of the plant (e.g., switchyard) is sufficiently low compared to LOOP and fire events already included in the internal events and fire PRAs. Therefore, they will not significantly impact the RICT Program calculations and can be excluded from RICT Program evaluation.

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Table E4-4
Evaluation of Risks from External Hazards

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
	the assumption that fewer than 23 helicopter flights per year operate from the Calvert Cliffs helipad, which is approximately 1500 feet from the containment and auxiliary buildings.		
External Flooding	The external flooding hazard at the site was recently updated as a result of the post-Fukushima Flood Hazard Reevaluation Request (FHRR) (Reference 21). The extreme flood hazards may challenge the plant, but measures have been put in place to protect the plant from the effects of the two flood hazards affecting the plant: local intense precipitation (LIP) and probable maximum storm surge (PMSS) (References 22 and 23). In addition to the compensatory measures, the following are taken into consideration screening the risk from these hazards: • LIP – The frequency of a LIP event that challenges the plant is estimated to be well below 10^{-6}/yr (Reference 22). • PMSS – A hurricane is the source of the PMSS. Due to the location of	Loss of offsite power and/or loss of SW are potential challenges within the design basis.	Weather-related LOOP events are accounted for in the internal events PRA. Loss of SW due to PMSS only occurs during a hurricane, for which the plant would be shutdown. Therefore, external flooding can be excluded from the RICT Program evaluation.

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Table E4-4
Evaluation of Risks from External Hazards

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
	CCNPP within the Chesapeake Bay, the hurricane track resulting in a PMSS event is very unique (Reference 21); although not quantified, the frequency is judged to be low. Further, plant procedures direct the plant be shut down within 8 hours of the arrival of a hurricane (References 13 and 24); this would result in effectively exiting the RICT.		
Extreme Winds and Tornados (including generated missiles)	Tornadoes and tornado-generated missiles are treated using a bounding approach with change to RICT Program criteria (see Section 4). Although hurricanes may generate missiles also, procedures direct the plant to shut down within 8 hours of a hurricane strike (Reference 13). This results in entering conditions for which the RICT program does not apply. The impact of other high wind events is assumed to be a LOOP.	Loss of offsite power from extreme winds & tornadoes is a potential challenge within the design basis.	Weather-related LOOP is included in the internal events PRA, and accounts for the high wind (non-tornadic) events. During hurricanes, the plant would be shutdown and the RICT Program does not apply. Tornado impacts are accounted for as described in Attachment 2.
Turbine-Generated Missiles	The probabilistic analysis performed for failures of turbines in Units 1 & 2 shows that the core damage risk associated with turbine missiles is much less than $1x10^{-6}$ per year. The probability of turbine missile	There are no challenges presented to the CCNPP site from turbine generated missiles.	Excluded from RICT Program evaluation.

Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

Table E4-4
Evaluation of Risks from External Hazards

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
	damage is $7x10^{-8}$/year for Unit 1. The probability of turbine missile damage is not calculated for Unit 2, since the turbine missile generating frequency is $9x10^{-7}$/year for Unit 2 (i.e., less than 10^{-6} per year). Note that these values are calculating assuming that the turbine valves are tested quarterly. (Reference 7)		
External Fires	Fires due to transportation accidents, industrial facilities, and pipelines are discussed in their respective sections in this table. For forest fires, an area of woodlands surrounds the site on three sides. There is adequate clearing around the site to form a fire break. Although some smaller stands of trees are within 250 feet of the center of the site, the major forested areas are between 500 to 750 feet from the control room. Although not designed for protection from smoke, the control room HVAC system can be placed in the recirculation mode, which minimizes the use of outside air.	Fire in the vicinity of the plant could potentially result in a loss of offsite power (LOOP).	The potential LOOP due to a forest fire is accounted for in the internal events PRA. Otherwise, external fires are excluded from RICT Program evaluation.
Accidents From Nearby Facilities	The Dominion Cove Point Liquid Natural Gas (LNG) Terminal is located approximately 3½ miles from CCNPP. An analysis was performed in 2006 as part of a	There are no challenges presented to the CCNPP site from accidents at nearby facilities.	Excluded from RICT Program evaluation.

Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

Table E4-4
Evaluation of Risks from External Hazards

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
	proposed expansion of the terminal and its operations (Reference 20). Previous studies (e.g., Reference 25) concluded that the risk to CCNPP was negligible. The updated study estimates the risk of LNG operations contained within the area, including LNG ships enroute, berthing of ships and cargo transfer, storage and processing at the onshore facility, and pipeline export. The risk of fatality at CCNPP and the risk of physical damage to the plant was estimated at significantly less than 10-6/yr. There are no other substantial industrial or military facilities within 5 miles of CCNPP (Reference 26).		
Release of Chemicals Stored at the Site	A separate hazard analysis was performed as part of the IPEEE (Reference 7). The analysis screened and/or evaluated the probability of a release which could result in a loss of Control Room habitability, incapacitation of operators, damage to vital equipment and subsequent off site exposure levels exceeding 10CFR100 limits. It was concluded that no chemicals	There are no challenges presented to the CCNPP site from chemicals stored onsite.	Excluded from RICT Program evaluation.

Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

Table E4-4
Evaluation of Risks from External Hazards

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
	stored onsite posed a significant hazard to Control Room operability or plant equipment.		
Transportation Accidents	Analysis of accidents on transportation routes (other than airways) in the vicinity of CCNPP was performed in the IPEEE (Reference 7). Additionally, a more recent study performed for Calvert Cliffs Unit 3 (Reference 26) showed that transportation accidents (other than aircraft) represented a negligible risk to CCNPP. Aircraft accidents are discussed in a previous part of this table.	There are no challenges presented to the CCNPP site from transportation accidents.	Excluded from RICT Program evaluation.
Pipeline Accidents (e.g., natural gas)	A pipeline from the Dominion Cove Point LNG Terminal runs through the center of Calvert County and passes about 2 miles from CCNPP. The study (Reference 20) performed for the LNG terminal includes hazards from this pipeline. Based on the study, the pipeline accident hazard to the site is much less than 10^{-6}/yr. There are no other known pipelines in the vicinity of CCNPP.	There are no challenges presented to CCNPP as a result of pipeline accidents.	Excluded from RICT Program evaluation.

Table E4-5 Seismic LOOP Frequency Estimate

Acceleration (g)	Seismic Interval (g)	Interval Representative g Level	Interval Frequency (/yr)	Offsite Power Failure Prob.	Weighted Average LOSP freq
0.05	0.05-0.075	0.06	1.63E-04	1.65E-03	2.69E-07
0.075	0.075-0.15	0.11	9.16E-05	2.73E-02	2.50E-06
0.15	0.15-0.30	0.21	2.10E-05	2.61E-01	5.48E-06
0.3	0.30-0.50	0.39	3.88E-06	6.82E-01	2.64E-06
0.5	0.50-0.70	0.59	8.05E-07	8.95E-01	7.21E-07
0.7	0.70-0.90	0.79	3.73E-07	9.64E-01	3.59E-07
0.9	0.90-1.1	0.99	1.46E-07	9.87E-01	1.44E-07
1.1	>1.1	1.5	1.87E-07	9.99E-01	1.86E-07
Total Seismic LOSP Frequency =					1.2E-05

6. Conclusions

Based on this analysis of external hazards for CCNPP Units 1 and 2, no additional external hazards other than seismic events and tornadoes need to be added to the existing PRA model. The evaluation concluded that the hazards either do not present a design-basis challenge to CCNPP, the challenge is adequately addressed in the PRA, or the hazard has a negligible impact on the calculated RICT and can be excluded.

The ICDP/ILERP acceptance criteria of 1E-5/1E-6 will be used within the PARAGON framework to calculate the resulting RICT and RMAT based on the total configuration-specific delta CDF/LERF attributed to internal events and internal fire, plus the seismic and tornado risk bounding delta CDF/LERF values.

7. References

1. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

2. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238).

3. WCAP-16952-NP, "Supplemental Implementation Guidance for the Calculation of Risk Informed Completion Time and Risk Managed Action Time for RITSTF Initiative 4B," August 2010.

4. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk- Informed Decision Making," Volume 1, March 2009.

5. NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," 1975.

6. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

7. "CCNPP Individual Plant Examination External Events," RAN 97-031.

8. Lettis Consultants International (LCI), Inc., Project No. 1041, Calvert Cliffs Hazard and Screening Report, 2013.

9. "Seismic Hazard and Screening Report in Response to the 50.54(I) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for the Calvert Cliffs Nuclear Power Plant", dated March 12, 2014, (S&L Report SL-012325, Revision 0); the relevant hazard curves in that report are taken from Reference 2.

10. Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," U.S. NRC IN 2010-18, September 2, 2010.

11. Staff Report, "Implications of Updated Probabilistic Seismic Hazard Estimates In Central And Eastern United States On Existing Plants, Safety/Risk Assessment," ML100270639, August 2010.

12. "CCNPP Units 1 and 2 Quantification Notebook, 2015 FPIE Model Update [at power, internal events model revision CA015A]," Calculation No. CA-PRA-014, Revision 2, November, 2015.

13. "Immediate Actions," CCNPP Technical Procedure, ERPIP-3.0, Revision 05800.

14. CCNPP UFSAR, Revision 47.

15. Letter from NRC to R.E. Denton (BGE), "Licensing Basis Charge (sic) for Evaluating the Need for Tornado Generated Missile Barriers – Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90621 and M90622)," dated May 1, 1995.

16. CCNPP PRA Evaluation Request #C0-2012-022, October 9, 2012.

17. NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," Revision 2, US Nuclear Regulatory Commission, February 2007.

18. CA-LAR-004, "High Winds Analysis," Revision 0.

Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

19. "CCNPP Aircraft Hazard Analysis," RAN: 97-034B1, Revision 5, February 22, 2002.

20. Letter from J.A. Spina (Constellation Energy) to NRC, "Calvert Cliffs Nuclear Power Plant; Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318; Independent Spent Fuel Storage Installation; Docket No. 72-8; Revision to Hazards Analysis Related to Liquefied Natural Gas Plant Operations at Cove Point," February 20, 2008 [ADAMS ML080560423].

21. Letter from NRC to G.H. Gerlich (Exelon), "Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Staff Assessment of Response to 10 CFR 50.54(f) Information Request – Flood-causing Mechanism Reevaluation (TAC NOS. MF3097 and MF3098)," April 16, 2015 [ADAMS ML15077A103].

22. Calvert Cliffs Condition Report CR-2013-001914, "Fukushima Project: Per CCNPP Flood Hazard Re-evaluation Repo," March 1, 2013.

23. Calvert Cliffs Condition Report CR-2013-001913, "Fukushima Project: Per CCNPP Flood Hazard Re-evaluation Repo," February 28, 2013.

24. "Severe Weather Preparation," CCNPP Station Administrative Procedure EP-1-108, Revision 01400.

25. Letter from Mr. R. E. Denton (BGE) to Document Control Desk (NRC), dated June 7, 1993, "Liquefied Natural Gas Hazards Analysis."

26. CCNPP Unit 3 Combined Licensed Application, Part 2: Final Safety Analysis Report, Revision 10, September 2014.

27. "Risk Assessment of Operational Events, Volume 2 – External Events – Internal Fires – Internal Flooding – Seismic – Other External Events – Frequencies of Seismically-Induced LOOP Events (RASP Handbook)", Revision 1.01, US Nuclear Regulatory Commission, January 2008.

28. CCNPP EDG AOT extension submittal, NRC ADAMS accession number ML031360410, Calvert Cliffs document RAN-02-020 Revision 2.

ENCLOSURE 5

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

1. Introduction

Section 4.0, Item 6 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide the plant-specific total CDF and LERF to confirm applicability of the limits of Regulatory Guide (RG) 1.174, Revision 1 (Reference 3). (Note that RG 1.174, Revision 2 (Reference 4), issued by the NRC in May 2011, did not revise these limits.)

The purpose of this enclosure is to demonstrate that the CCNPP total Core Damage Frequency (CDF) and total Large Early Release Frequency (LERF) are below the guidelines established in RG 1.174. RG 1.174 does not establish firm limits for total CDF and LERF, but recommends that risk-informed applications be implemented only when the total plant risk is no more than about 1E-4/year for CDF and 1E-5/year for LERF. Demonstrating that these limits are met confirms that the risk metrics of NEI-06-09 can be applied to the Calvert Cliffs Nuclear Power Plant (CCNPP) Risk Informed Completion Time (RICT) Program.

2. Technical Approach

Table E5-1 lists the CCNPP Unit 1 and Unit 2 CDF and LERF values that resulted from a quantification of the baseline internal events (including internal flooding) and fire Probabilistic Risk Assessment (PRA) models (References 5 and 6, respectively). This table also includes an estimate of the seismic and high winds contribution to CDF and LERF (References 7 and 8, respectively). The seismic and high winds CDF/LERF are based on the methodology detailed in Enclosure 4. Other external hazards are below accepted screening criteria and therefore do not contribute significantly to the totals.

Table E5-1
Total Baseline CDF/LERF

Unit 1 Baseline CDF			**Unit 1 Baseline LERF**	
Source	**Contribution**		**Source**	**Contribution**
Internal Events PRA	9.5E-06		Internal Events PRA	1.2E-06
Fire PRA	4.2E-05*		Fire PRA	3.2E-06*
Seismic	1.1E-06		Seismic	1.1E-07
High Winds	3.3E-07		High Winds	1.6E-08
Other External Events	No significant contribution		Other External Events	No significant contribution
Total Unit 1 CDF	**5.3E-05**		**Total Unit 1 LERF**	**4.5E-06**

Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

Table E5-1
Total Baseline CDF/LERF

Unit 2 Baseline CDF	
Source	**Contribution**
Internal Events PRA	9.6E-06
Fire PRA	4.0E-05*
Seismic	1.1E-06
High Winds	5.4E-07
Other External Events	No significant contribution
Total Unit 2 CDF	**5.1E-05**

Unit 2 Baseline LERF	
Source	**Contribution**
Internal Events PRA	1.2E-06
Fire PRA	3.4E-06*
Seismic	1.1E-07
High Winds	2.9E-08
Other External Events	No significant contribution
Total Unit 2 LERF	**4.8E-06**

*Note: The values of CDF and LERF for the Fire PRA provided in Table E5-1 reflect installation of the modifications listed in Attachment 5.

As demonstrated in Table E5-1, the total CDF and total LERF are within the guidelines set forth in RG 1.174, and support small changes in risk that may occur during RICT entries following TSTF-505 implementation. Therefore, CCNPP TSTF-505 implementation is consistent with NEI 06-09 guidance.

3. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238).

2. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

3. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

4. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011 (Accession No. ML10091006).

5. Calvert Cliffs Nuclear Power Plant, Units 1 and 2, PRA Model, CA015A.

6. Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Fire Risk Evaluation Report, CO-FRE-F001, Attachment 3: Risk Results Summary, Revision 2, February 2016.

Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

7. CA-LAR-003 R00, Bounding Seismic CDF and LERF Estimate for TSTF-505 (RICT) Program, November 2015.

8. CA-LAR-004 R00, High Winds Risk Analysis in Support of CCNPP 4b Submittal, November 2015.

ENCLOSURE 6

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Justification of Application of At-Power PRA Models to Shutdown Modes

This enclosure is not applicable to the Calvert Cliffs Nuclear Power Plant submittal. Exelon is proposing to apply the Risk-Informed Completion Time Program only in Modes 1 and 2 and not in the shutdown Modes.

ENCLOSURE 7

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

<u>PRA Model Update Process</u>

PRA Model Update Process

1. Introduction

Section 4.0, Item 8 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide a discussion of the licensee's programs and procedures which assure the PRA models which support the RMTS are maintained consistent with the as-built/as-operated plant.

This enclosure describes the administrative controls and procedural processes applicable to the configuration control of PRA models used to support the Risk-Informed Completion Time (RICT) Program, which will be in place to ensure that these models reflect the as-built/as-operated plant. Plant changes, including physical modifications and procedure revisions, will be identified and reviewed prior to implementation to determine if they could impact the PRA models per ER-AA-600-1015, FPIE [Full Power Internal Events] PRA Model Update (Reference 3), and ER-AA-600-1061, Fire PRA Model Update and Control (Reference 4). The configuration control program will ensure these plant changes are incorporated into the PRA models as appropriate. The process will include discovered conditions associated with the PRA models, which will be addressed by the applicable site Corrective Action Program.

Should a plant change or a discovered condition be identified that has a significant impact to the RICT Program calculations as defined by the above procedures, an unscheduled update of the PRA model will be implemented. Otherwise, the PRA model change is incorporated into a subsequent periodic model update. Such pending changes are considered when evaluating other changes until they are fully implemented into the PRA models. Periodic updates are typically performed every four years.

2. PRA Model Update Process

Internal Event, Internal Flood, and Fire PRA Model Maintenance and Update

The Fleet risk management process ensures that the applicable PRA model used for the RICT Program reflects the as-built/as-operated plant for each of the Calvert Cliffs units. The PRA configuration control process delineates the responsibilities and guidelines for updating the full power internal events, internal flood, and fire PRA models, and includes both periodic and unscheduled PRA model updates.

The process includes provisions for monitoring potential impact areas affecting the technical elements of the PRA models (e.g., due to plant changes, plant/industry operational experience, or errors or limitations identified in the model), assessing the individual and cumulative risk impact of unincorporated changes, and controlling the model and necessary computer files, including those associated with the configuration risk management program (CRMP) model.

Review of Plant Changes for Incorporation into the PRA Model

1. Plant changes or discovered conditions are reviewed for potential impact to the PRA models, including the CRMP model and the subsequent risk calculations which support

the RICT Program (NEI 06-09, Section 2.3.4, Items 7.2 and 7.3, and 2.3.5, Items 9.2 and 9.3).

2. Plant changes that meet the criteria defined in References 3 and 4 (including consideration of the cumulative impact of other pending changes) will be incorporated in the applicable PRA model(s), consistent with the NEI 06-09 guidance. Otherwise, the change is assigned a priority and is incorporated at a subsequent periodic update consistent with procedural requirements. (NEI 06-09, Section 2.3.5, Item 9.2)

3. PRA updates for plant changes are performed at least once every two refueling cycles, consistent with the guidance of NEI 06-09 (NEI 06-09, Section 2.3.4, Item 7.1, and 2.3.5, Item 9.1).

4. If a PRA model change is required for the CRMP model, but cannot be immediately implemented for a significant plant change or discovered condition, either:

   a. Interim analyses to address the expected risk impact of the change will be performed. In such a case, these interim analyses become part of the RICT Program calculation process until the plant changes are incorporated into the PRA model during the next update. The use of such bounding analyses is consistent with the guidance of NEI 06-09.

   b. Appropriate administrative restrictions on the use of the RICT Program for extended Completion Times are put in place until the model changes are completed, consistent with the guidance of NEI 06-09.

   These actions satisfy NEI 06-09, Section 2.3.5, Item 9.3.

3. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238).

2. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

3. ER-AA-600-1015, "FPIE PRA Model Update."

4. ER-AA-600-1061, "Fire PRA Model Update and Control."

ENCLOSURE 8

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

<u>Attributes of the CRMP Model</u>

Attributes of the CRMP Model

1. Introduction

Section 4.0, Item 9 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide a description of PRA models and tools, including identification of how the baseline probabilistic risk assessment (PRA) model is modified for use in the configuration risk management program (CRMP) tools, quality requirements applied to the PRA models and CRMP tools, consistency of calculated results from the PRA model and the CRMP tools, and training and qualification programs applicable to personnel responsible for development and use of the CRMP tools. This item should also confirm that the CRMP tools can be readily applied for each Technical Specification (TS) limiting condition for operation (LCO) within the scope of the plant-specific submittal.

This enclosure describes the necessary changes to the peer-reviewed baseline PRA models for use in the CRMP software to support the Risk-Informed Completion Time (RICT) Program. The process employed to adapt the baseline models for CRMP use is demonstrated:

a) to preserve the core damage frequency (CDF) and large early release frequency (LERF) quantitative results;
b) to maintain the quality of the peer-reviewed PRA models; and
c) to correctly accommodate changes in risk due to configuration-specific considerations.

Quality controls and training programs applicable for the CRMP are also discussed in this enclosure.

2. Translation of Baseline PRA Model for Use in CRMP

The baseline PRA models for internal events, including the internal flood and internal fire models, are the peer-reviewed models, updated when necessary to incorporate plant changes to reflect the as-built/as-operated plant. These models will be used in the RICT Program.

The CRMP software will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. Maintenance alignment probabilities in the baseline PRA models have probabilities based on the fraction of the year the equipment is unavailable. For the CRMP model, the actual configuration of equipment is evaluated, so the maintenance alignment probabilities are set to zero.

The current Calvert Cliffs core design reflected in the baseline PRA model for ATWS events includes a UET (Unfavorable Exposure Time) for variable success criteria based on time of core life (i.e., moderator temperature coefficient early in cycle life). The event is set to the fraction of the year for which the UET applies, and will be changed to a probability of 1 or 0 based on the actual time in the operating cycle.

Attributes of the CRMP Model

3. Quality Requirements and Consistency of PRA Model and CRMP Tools

The approach for establishing and maintaining the quality of the PRA models, including the CRMP model, includes both a PRA maintenance and update process (described in Enclosure 7), and the use of self-assessments and independent peer reviews (described in Enclosure 2).

The information provided in Enclosure 2 demonstrates that the site's internal event, internal flood, and internal fire PRA models reasonably conform to the associated industry standards endorsed by Regulatory Guide 1.200 (Reference 3). This information provides a robust basis for concluding that the PRA models are of sufficient quality for use in risk-informed licensing actions.

For maintenance of an existing CRMP model, changes made to the baseline PRA model in translation to the CRMP model will be controlled and documented. An acceptance test is performed after every CRMP model update. This testing also verifies correct mapping of plant components to the basic events in the CRMP model.

4. Training and Qualification

The PRA staff is responsible for development and maintenance of the CRMP model. Operations and Work Control staff will use the CRMP tool under the RICT Program. PRA Staff and Operations are trained in accordance with a program using National Academy for Nuclear Training (ACAD) documents, which is also accredited by INPO.

5. Application of the CRMP Tool to the RICT Program Scope

The PARAGON software will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. This program is specifically designed to support implementation of RMTS. PARAGON will permit the user to evaluate all configurations within the scope of the RICT Program using appropriate mapping of equipment to PRA basic events. The RICT program will meet RG 1.174 (Reference 4) and Exelon software quality assurance requirements.

6. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238).

2. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

Attributes of the CRMP Model

3. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

4. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011.

ENCLOSURE 9

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Key Assumptions and Sources of Uncertainty

Key Assumptions and Sources of Uncertainty

1. Introduction

The purpose of this enclosure is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for the Risk Informed Completion Time (RICT) Program. Topical Report NEI 06-09 (Reference 1), Section 2.3.4, item 10 requires an evaluation to determine insights that will be used to develop risk management actions (RMAs) to address these uncertainties. The baseline internal events PRA and fire PRA (FPRA) models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to the RICT Program calculations, to perform sensitivity analyses where appropriate, to discuss the results and to provide dispositions for the RICT Program.

The epistemic uncertainty analysis approach described below applies to the internal events PRA and any epistemic uncertainty impacts that are unique to FPRA are also addressed. In addition, Topical Report NEI 06-09 requires that the uncertainty be addressed in RICT Program Configuration Risk Management Program (CRMP) tools by consideration of the translation from the PRA model to the CRMP tool. The CRMP model, also referred to as the PARAGON model, discussed in Enclosure 6 is an integrated model representing internal events, flooding events and fire events. The model translation uncertainties evaluation and impact assessment are limited to new uncertainties that could be introduced by application of the CRMP tool during RICT Program calculations.

2. Assessment of Internal Events PRA Epistemic Uncertainty Impacts

In order to identify key sources of uncertainty for RICT Program application, the internal events baseline PRA model uncertainty report was developed, based on the guidance in NUREG-1855 (Reference 2) and EPRI 1016737 (Reference 3). As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the Calvert Cliffs Nuclear Power Plant (CCNPP) baseline PRA model quantification (Reference 4).

Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the CCNPP internal events PRA technical elements are noted in the individual notebooks. These assumptions were collected from each notebook. The internal events PRA model uncertainties evaluation is documented in Reference 5, and considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. The Electric Power Research Institute (EPRI) compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (Reference 3), and the evaluation performed for CCNPP (Reference 5) considered each of the generic sources of modeling uncertainty as well as the plant-specific sources.

Key Assumptions and Sources of Uncertainty

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application (Reference 4). No specific issues of PRA completeness have been identified relative to the TSTF-505 application, based on the results of the internal events PRA and fire PRA peer reviews.

Based on the review of sources of uncertainty, the following items were identified for evaluation as potential key sources of uncertainty for the RICT application.

- Human error probabilities (HEPs) in the internal events PRA
- Internal flood PRA initiating event frequency methodology

These are discussed in Table E9-1.

Key Assumptions and Sources of Uncertainty

Table E9-1
Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Uncertainties associated with the assumptions and method of calculation of HEPs for the Human Reliability Analysis (HRA) may introduce uncertainty. Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	Potentially all LCOs in the RICT program	Sensitivity cases for the base internal events PRA (use of 5th and 95th percentile value HEPs) show that the results are somewhat sensitive to HRA model and parameter values. The CCNPP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. The model results assuming 95th percentile values indicate some sensitivity to human performance. Use of 95th percentile HEPs is not considered realistic given the consistent application of a consensus HRA approach. However, the TSTF-505 procedure will require appropriate risk management action (RMA) focus on human performance for RICT entry, e.g., including an operator briefing on the significant human actions in the PRA that are pertinent to the configuration. Refer to Enclosure 12 for additional discussion on RMAs.

Key Assumptions and Sources of Uncertainty

Table E9-1 Assessment of Internal Events PRA Epistemic Uncertainty Impacts		
Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Some of the pipe rupture frequencies in the internal flooding PRA are based on an older modeling approach. Conversion to the newer method may increase internal flood CDF. Portions of the internal flood model use a pipe segment-based method to determine pipe rupture frequencies, whereas more recently-updated portions of the model use the newer pipe length approach per EPRI TR-1013141 (Ref. 6)	LCOs for which internal flood risk may have a significant effect on RICT	Prior to implementation of the RICT program, the internal flood model will be updated so that the model consistently uses the newer methodology. Therefore, this uncertainty will not be an issue for RICT calculations.

3. Assessment of Translation (CRMP Model) Uncertainty Impacts

Incorporation of the baseline PRA models into the CRMP model used for RICT Program calculations may introduce new sources of model uncertainty. Table E9-2 provides a description of the relevant model changes and dispositions of whether any of the changes made represent possible new sources of model uncertainty that must be addressed.

Key Assumptions and Sources of Uncertainty

Table E9-2
Assessment of Translation Uncertainty Impacts

CRMP Model Change and Assumptions	Part of Model Affected	Impact on Model	Disposition
PRA model logic structure may be optimized to increase solution speed.	Fault tree logic model structure, affecting both internal and fire PRAs	The model, if restructured, will be logically equivalent and produce results comparable to the baseline PRA logic model	Since the restructured model will produce comparable numerical results, this is not a source of uncertainty for the RICT program.
Incorporation of seismic and high wind risk bias to support RICT Program risk calculations. A conservative value for both the seismic and high wind delta CDF is applicable.	Calculation of RICT and RMAT within CRMP	The addition of bounding impacts for seismic and high wind events has no impact on baseline PRA or CRMP model. Impact is reflected in calculation of all RICTs and RMATs.	Since this is a bounding approach for addressing seismic and high wind risks in the RICT Program, it is not a source of uncertainty, and RICT Program calculations are not impacted, so no mandatory RMAs are required.
Set plant availability (Reactor Critical Years Factor) basic event to 1.0.	Basic event RCF	Since the CRMP model evaluates specific configurations during at-power conditions, the use of a plant availability factor less than 1.0 is not appropriate. This change allows the CRMP model to produce appropriate results for specific at-power configurations.	This change is consistent with CRMP tool practice; therefore this change does not represent a source of uncertainty, and RICT program calculations are not impacted, so no mandatory RMAs are required.

Key Assumptions and Sources of Uncertainty

4. Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

The purpose of the following discussion is to address the epistemic uncertainty in the CCNPP FPRA. The CCNPP FPRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the FPRA and because the state of knowledge in these elements continues to evolve. The development of the CCNPP FPRA was guided by NUREG/CR-6850 (Reference 7). The CCNPP FPRA model used consensus models described in NUREG/CR-6850. Enclosure 2 provides a detailed discussion of all the Peer Review F&Os and the resolutions.

CCNPP used guidance provided in NUREG/CR-6850 (Reference 7) and NUREG-1855 (Reference 2) to address uncertainties associated with FPRA for the RICT Program application. As stated in Section 1.5 of NUREG-1855:

> "Although the guidance does not currently address all sources of uncertainty, the guidance provided on the process for their identification and characterization and for how to factor the results into the decision making is generic and is independent of the specific source. Consequently, the process is applicable for other sources such as internal fire, external events, and low power and shutdown."

NUREG-1855 also describes an approach for addressing sources of model uncertainty and related assumptions. It defines:

> "A source of model uncertainty is one that is related to an issue in which no consensus approach or model exists and where the choice of approach or model is known to have an effect on the PRA (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion and introduction of a new initiating event)."

NUREG-1855 defines consensus model as:

> "A model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that NRC has utilized or accepted for the specific risk-informed application for which it is proposed."

The potential sources of model uncertainty in the CCNPP FPRA model were characterized in Reference 8 for the 16 tasks identified by NUREG/CR-6850. This framework was used to organize the assessment of baseline FPRA epistemic uncertainty and evaluate the impact of this uncertainty on RICT Program calculations. Table E9-3 outlines sources of uncertainties by task and their disposition. The results of this assessment reflect the sensitivity analyses that have been performed.

As noted above, the CCNPP FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC. Further,

Key Assumptions and Sources of Uncertainty

appropriate cable impacts were identified for the systems modeled in the Internal Events PRA and were modeled in the Fire PRA. No systems were conservatively assumed to be failed for all FPRA scenarios. All Fire PRA methods were based on NUREG/CR-6850 and published "frequently asked questions" (FAQs) for the FPRA.

In addition to the discussion of sources of model uncertainty in Table E9-3, the evaluation of sources of model uncertainty in the FPRA and associated sensitivity studies identified two modeling uncertainties that may be potentially significant for applications. These are:

- Reliability of the Halon system for specific locations such as the Cable Spreading Rooms and the Switchgear Rooms
- Human error probabilities in the fire PRA
- Assumptions regarding impact of transient fires

These are addressed in Table E9-4.

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Table E9-3 Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
1	Analysis boundary and partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on the discussion of sources of uncertainly it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
2	Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized through use of the PWROG Generic MSO list and the process used to identify and assess potential MSOs. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3 Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
3	Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Based on the discussion of sources of uncertainty it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
4	Qualitative Screening	Qualitative screening was performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

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Table E9-3 Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
5	Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology used is consistent with that used for the internal events PRA model development as was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process would have reviewed all significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3 Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
6	Fire Ignition Frequency	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates. CCNPP currently uses the NUREG/CR-6850 Supplement 1 (Reference 8) ignition frequencies. The fire frequency values are believed to currently be over-estimated. A future model update will address the new ignition frequencies in NUREG-2169 (Reference 9) along with the recently revised heat release rates from NUREG 2178 (Reference 10). This is considered to be part of the normal FPRA model maintenance process.	Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would require sensitivity treatment. Consensus approaches are employed in the current model and will be employed as appropriate in future model updates. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

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Table E9-3 Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
7	Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The CCNPP FPRA development did not screen out any fire initiating events based on low CDF/LERF contribution. Screening of individual fire ignition sources occurred only if it involved a discrete component and the consequences of the associated fire did not involve failure of any other plant component or feature. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Quantitative Screening task *does not introduce any epistemic uncertainties* that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
8	Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are 8 and 11. The discussion of uncertainty for both tasks is provided in the discussion for Task 11.	See Task 11 discussion.

Key Assumptions and Sources of Uncertainty

Table E9-3 Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
9	Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG 7150, Volume 2 (Reference 11), based on actual fire test data, were used in the CCNPP Fire PRA. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3 Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
10	Circuit Failure Mode Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG 7150, Volume 2 (Reference 11). The uncertainty values specified in NUREG 7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG 7150, Volume 2. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
11	Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution	Consensus modeling approach is used for the Detailed Fire Modeling. Except as noted in Table E9-4, it is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no additional RMAs are required to address this item.

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Table E9-3 Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition
		(aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	

Key Assumptions and Sources of Uncertainty

Table E9-3 Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
12	Post-Fire Human Reliability Analysis	The human error probabilities used in the FPRA were adjusted to consider the additional challenges that may be present given a fire. The human error probabilities were obtained using the EPRI HRAC and included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The human error probabilities were obtained using the EPRI HRAC and included the consideration of degradation or loss of necessary cues due to fire. The impact of any remaining uncertainties is expected to be small. Except as noted in Table E9-4, it is concluded that the methodology for the Post-Fire Human Reliability Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no additional RMAs are required to address this item.
13	Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified FPRA model. A conservative seismic hazard penalty is already applied to all RICT calculations to account for seismic risk impact. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

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Table E9-3 Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
14	Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit. However, the selected truncation was confirmed to be consistent with the requirements of the PRA Standard.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
15	Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

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Table E9-3 Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition
16	FPRA Documentation	This task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines *documentation requirements*. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the FPRA documentation task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-4
Treatment of Specific Fire PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Sensitivities performed for FPRA sources of fire modeling uncertainty indicate that assumptions regarding Halon (fire suppression) system reliability for cable spreading room (CSR) and switchgear room (SWGR) scenarios are important to the FPRA results.	LCOs associated with configurations affecting automatic fire suppression; LCOs associated with configurations in which CSR or SWGR fire sequences are important CDF/LERF contributors	Consideration should be given to appropriate risk management actions for the associated equipment and fire areas consistent with the CRMP. Refer to Enclosure 12 for additional discussion on RMAs.
Uncertainties associated with the assumptions and method of calculation of HEPs for the Human Reliability Analysis (HRA) may introduce uncertainty. Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	Potentially all LCOs in the RICT program	Sensitivity cases for the base fire PRA (assuming all HEPs = 1.0 or 0.0) show that the results are somewhat sensitive to HRA model and parameter values. The CCNPP FPRA model HRA is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Assuming no credit for operator response is not realistic. However, the TSTF-505 procedure will require appropriate risk management action (RMA) focus on human performance for RICT entry, e.g., including an operator briefing on the significant human actions in the PRA that are pertinent to the configuration. Refer to Enclosure 12 for additional discussion on RMAs.

Key Assumptions and Sources of Uncertainty

Table E9-4
Treatment of Specific Fire PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Assumptions regarding impact of transient fires may introduce uncertainty.	LCOs associated with configurations in which the impact of transients is important to FPRA CDF/LERF results.	Consideration should be given to appropriate risk management actions, e.g., to limit transient combustibles and hot work in fire areas that are important to the configuration-specific CDF/LERF results, consistent with the CRMP. Refer to Enclosure 12 for additional discussion on RMAs.

5. References

1. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

2. NUREG-1855, "Guidance on the treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Main Report, March 2009.

3. EPRI TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," Electric Power Research Institute, Final Report, December 2008.

4. CA-PRA-014, Revision 2, "Calvert Cliffs PRA Quantification Notebook, Units 1 and 2," November 2015.

5. CA-PRA-026, Rev. 0, "Calvert Cliffs Uncertainty Assessment Notebook", November 2015.

6. EPRI TR-1013141, "Pipe Rupture Frequencies for Internal Flooding PRAs," Rev. 1, Electric Power Research Institute, Palo Alto, CA, 2006.

7. NUREG/CR-6850 (also EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," September 2005, with Supplement 1 (EPRI 1019259), September 2010.

8. C0-UNC-001, "Calvert Cliffs Fire PRA Uncertainty and Sensitivity Analysis Notebook," Revision 1, August 2013.

9. "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009," NUREG-2169/EPRI 3002002936, U.S. NRC and Electric Power Research Institute, January 2015.

10. "Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume," NUREG-2178 Vol. 1/ EPRI 3002005578, U.S. NRC and Electric Power Research Institute, Draft Report for Comment, April 2015.

11. "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," Final Report, NUREG/CR-7150, Vol. 1, EPRI 3002001989, U.S. NRC and Electric Power Research Institute, May 2014.

ENCLOSURE 10

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

<u>Program Implementation</u>

Program Implementation

1. Introduction

Section 4.0, Item 11 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0-A (Reference 2) requires that the license amendment request (LAR) provide a description of the implementing programs and procedures regarding the plant staff responsibilities for the Risk Managed Technical Specifications (RMTS) implementation, and specifically discuss the decision process for risk management action (RMA) implementation during a Risk-Informed Completion Time (RICT).

This enclosure provides a description of the implementing programs and procedures regarding the plant staff responsibilities for the Risk-Informed Completion Time (RICT) Program, including training of plant personnel, and specifically discusses the decision process for RMA implementation during extended Completion Times (CT).

2. RICT Program and Procedures

Exelon will develop a program description and implementing procedures for the RICT Program. The program description will establish the management responsibilities and general requirements for risk management, training, implementation, and monitoring of the RICT program. More detailed procedures will provide specific responsibilities, limitations, and instructions for implementing the RICT program. The program description and implementing procedures will incorporate the programmatic requirements for RMTS included in NEI 06-09.

The Operations Department (licensed operators) is responsible for compliance with the TS and will be responsible for implementation of RICTs and RMAs. Entry into the RICT program will require management approval prior to pre-planned activities and as soon as practicable following emergent conditions.

The procedures for the RICT program will address the following attributes consistent with NEI 06-09:

- Plant management positions with authority to approve entry into the RICT Program.
- Important definitions related to the RICT Program.
- Departmental and position responsibilities for activities in the RICT Program.
- Plant conditions for which the RICT Program is applicable.
- Limitations on implementing RICTs under voluntary and emergent conditions.
- Implementation of the RICT Program 30-day back stop limit.
- Use of the Configuration Risk Management Program (CRMP) tool.
- Guidance on recalculating RICT and risk management action time (RMAT) within 12 hours or within the most limiting front-stop CT after a plant configuration change.
- Requirements to identify and implement RMAs when the RMAT is exceeded or is anticipated to be exceeded.
- Guidance on the use of RMAs including the conditions under which they may be credited in RICT calculations.
- Guidance on crediting probabilistic risk assessment (PRA) functionality.

Program Implementation

- Conditions for exiting a RICT.
- Requirements for training on the RICT Program.
- Documentation requirements related to individual RICT evaluations, implementation of extended CTs, and accumulated annual risk.

The implementing procedures for the RICT program will also include special considerations associated with a total loss of TS-specified safety function condition. If a total loss of TS-specified safety function technical specification is emergently entered that allows a RICT, then program-defined actions will be taken in parallel with actions to restore the function to operability. The response taken will include protecting key equipment, addressing fire insights (e.g., Maintenance Rule fire risk), and external hazard considerations such as severe weather, with consideration of loss of offsite power as well as minimizing activities that could cause a plant transient. Operator briefings on relevant response procedures would be identified promptly and executed to maximize the ability to mitigate a transient in this emergent condition. Limitations on other work to minimize configuration risk would be imposed via the site online 10 CFR Part 50.65(a)(4) process to maximize remaining defense-in-depth.

Operational controls will drive the site to exit the loss of function state swiftly to ensure continued safe plant operation. Site management will be involved in the decision-making process including the senior-licensed Shift Manager and Station Leadership. If the condition cannot be resolved by promptly restoring a train, then and only then will PRA functionality be applied to an inoperable train for the calculation of a RICT. The PRA functional determination will not credit alternative structures, systems or components (SSCs), i.e., SSCs different from those referenced in the TS for the inoperable equipment, nor will it use PRA model success criteria (e.g., parameters such as flow rates, temperature limits, component response times) different from design basis, in order to preserve adequate safety margin while in the loss of function condition.

The PRA functional determination will include a review of significant internal events cutsets to provide high confidence that none of the design basis accidents, as modeled in the internal events PRA, proceed directly to core damage or containment failure. The remaining capability of the system will be credited appropriately in the RICT evaluation by only crediting events and scenarios that the SSC is physically capable of supporting. Exelon does not intend to take advantage of the full calculated RICT (using PRA functionality) for any loss of TS-specified function condition. Therefore, additional administrative controls (e.g., shorter completion times such as 24-48 hours) will be considered to minimize the time in the total loss of function configuration.

3. RICT Program Training

The scope of training for the RICT Program will include rules for the new TS program, CRMP software, TS Actions included in the program, and procedures. This training will be conducted for the following Exelon personnel:

Program Implementation

Site Personnel

- Operations Director
- Operations Personnel (Licensed and Non-Licensed)
- Operations Training
- Outage Manager
- On-line Manager
- Planning and Scheduling Personnel
- Work Week Managers
- Regulatory Assurance Personnel
- Selected Maintenance Personnel
- Engineering
- Risk Management
- Other Selected Management

Corporate Personnel

- Operations Corporate Functional Area Manager
- Fleet Outages Corporate Functional Area Manager
- Licensing Management and Personnel
- Risk Management Personnel and Managers
- Training Management and Personnel
- Other Selected Management

Training will be carried out in accordance with Exelon training procedures and processes. These procedures were written based on the Institute of Nuclear Power Operations (INPO) Accreditation (ACAD) requirements, as developed and maintained by the National Academy for Nuclear Training. Exelon has planned three levels of training for implementation of the RICT Program. They are described below:

Level 1 Training

This is the most detailed training. It is intended for the individuals who will be directly involved in the implementation of the RICT Program. This level of training includes the following attributes:

- Specific training on the revised TS
- Record keeping requirements
- Case studies
- Hands-on experience with the CRMP tool for calculating RMAT and RICT
- Identifying appropriate RMAs
- Determining PRA functionality
- Common cause failure considerations
- Other detailed aspects of the RICT Program

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Level 2 Training

This training is applicable to plant management positions with authority to approve entry into the RICT Program, as well as supervisors, managers, and other personnel who will closely support RICT implementation. These individuals need a broad understanding of the purpose, concepts, and limitations of the RICT Program. Level 2 training is significantly more detailed than Level 3 training (described below), but it is different from Level 1 training in that hands-on time with the CRMP tool, case studies, and other specifics are not required.

Level 3 Training

This training is intended for the remaining personnel who require an awareness of the RICT Program. These employees need basic knowledge of RICT Program requirements and procedures. This training will cover RICT Program concepts that are important to disseminate throughout the organization.

4. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238).

2. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

ENCLOSURE 11

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Monitoring Program

Monitoring Program

1. Introduction

Section 4.0, Item 12 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09 Revision 0-A (Reference 2) requires that the license amendment request (LAR) provide a description of the implementation and monitoring program as described in Regulatory Guide (RG) 1.174, An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1, (Reference 3) and NEI 06-09-A. (Note that RG 1.174, Revision 2 (Reference 4), issued by the NRC in May 2011, made editorial changes to the applicable section referenced in the NRC safety evaluation for Section 4.0, Item 12.)

This enclosure provides a description of the process applied to monitor the cumulative risk impact of implementation of the Risk-Informed Completion Time (RICT) Program, specifically the calculation of cumulative risk of extended Completion Times (CTs). Calculation of the cumulative risk for the RICT Program is discussed in Step 14 of Section 2.3.1 and Step 7.1 of Section 2.3.2 of NEI 06-09, *Risk Informed Technical Specifications Initiative 4b*. General requirements for a Performance Monitoring Program for risk-informed applications are discussed in Regulatory Guide (RG) 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis*, Element 3.

2. Description of Monitoring Program

The RICT Program will require calculation of cumulative risk impact at least every refueling cycle, not to exceed 24 months, consistent with the guidance in NEI 06-09, Revision 0-A. For the assessment period under evaluation, data will be collected for the risk increase associated with each application of an extended CT for both core damage frequency (CDF) and large early release frequency (LERF), and the total risk will be calculated by summing all risk associated with each RICT application. This summation is the change in CDF or LERF above the zero maintenance baseline levels during the period of operation in the extended CT (i.e., beyond the front-stop CT). The change in risk will be converted to average annual values.

The total average annual change in risk for extended CTs will be compared to the guidance of RG 1.174, Revision 2, Figures 4 and 5, acceptance guidelines for CDF and LERF, respectively. If the actual annual risk increase is acceptable (i.e., not in Region I of Figures 4 and 5 of RG 1.174, Revision 2), then RICT Program implementation is acceptable for the assessment period. Otherwise, further assessment of the cause of exceeding the acceptance guidelines of RG 1.174 and implementation of any necessary corrective actions to ensure future plant operation is within the guidelines will be conducted under the corrective action program.

The evaluation of cumulative risk will also identify areas for consideration, such as:

- RICT applications that dominated the risk increase
- Risk contributions from planned vs. emergent RICT applications
- Risk Management Actions (RMA) implemented but not credited in the risk
- calculations
- Risk impact from applying RICT to avoid multiple shorter duration outages
- Any specific RICT application that incurred a large proportion of the risk

Monitoring Program

Based on a review of the considerations above, corrective actions will be developed and implemented as appropriate. These actions may include:

- Administrative restrictions on the use of RICTs for specific high-risk configurations
- Additional RMAs for specific configurations
- Rescheduling planned maintenance activities
- Deferring planned maintenance to shutdown conditions
- Use of temporary equipment to replace out-of-service systems, structures, or
- components (SSC)
- Plant modifications to reduce risk impact of future planned maintenance configurations

In addition to impacting cumulative risk, implementation of the RICT Program may potentially impact the unavailability of SSCs. The existing Maintenance Rule (MR) monitoring programs under 10 CFR 50.65(a)(1) and (a)(2) provide for evaluation and disposition of unavailability impacts which may be incurred from implementation of the RICT Program. The SSCs in the scope of the RICT Program are also in the scope of the MR, which allows the use of the MR Program. RG 1.177, *An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications* (Reference 5), Section 3.2, Maintenance Rule Control, discusses that the scope of evaluations required under the Maintenance Rule should include prior related TS changes, such as extension of CTs.

The monitoring program for the MR, along with the specific assessment of cumulative risk impact described above, serve as the Implementation and Monitoring Program for the RICT Program as described in Element 3 of RG 1.174 and NEI 06-09.

3. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238).

2. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

3. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

4. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011.

5. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," Revision 1, May 2011.

ENCLOSURE 12

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

<u>Risk Management Action Examples</u>

Risk Management Action Examples

1. Introduction

This enclosure describes the process for identification and implementation of Risk Management Actions (RMA) applicable during extended Completion Times (CT) and provides examples of RMAs. RMAs will be governed by plant procedures for planning and scheduling maintenance activities. The procedures will provide guidance for the determination and implementation of RMAs when entering the Risk-Informed Completion Time (RICT) Program consistent with the guidance provided in NEI 06-09, Revision 0-A (Reference 1).

2. Responsibilities

For planned entries into the RICT Program, Work Management is responsible for developing the RMAs with assistance from Operations and Risk Management. Operations is responsible for approval and implementation of RMAs. For emergent entry into extended CTs, Operations is also responsible for developing the RMAs.

3. Procedural Guidance

For planned maintenance activities, implementation of RMAs will be required if it is anticipated that the risk management action time (RMAT) will be exceeded. For emergent activities, RMAs must be implemented if the RMAT is reached. Also, if an emergent event occurs requiring recalculation of a RMAT already in place, the procedure will require a reevaluation of the existing RMAs for the new plant configuration to determine if new RMAs are appropriate. These requirements of the RICT Program are consistent with the guidance of NEI 06-09.

RMAs will be implemented in accordance with current procedures (e.g., References 2, 3, 4, 5) no later than the time at which an incremental core damage probability (ICDP) of 1E-6 is reached, or no later than the time when an incremental large early release probability (ILERP) of 1E-7 is reached. If, as the result of an emergent condition, the instantaneous core damage frequency (ICDF) or the instantaneous large early release frequency (ILERF) exceeds 1E-3 per year or 1E-4 per year, respectively, RMAs are also required to be implemented. These requirements are consistent with the guidelines of NEI 06-09.

By determining which structures, systems, or components (SSCs) are most important from a CDF or LERF perspective for a specific plant configuration, RMAs may be created to protect these SSCs. Similarly, knowledge of the initiating event or sequence contribution to the configuration-specific CDF or LERF allows development of RMAs that enhance the capability to mitigate such events. The guidance in NUREG-1855 (Reference 6) and EPRI TR-1026511 (Reference 7) will be used in examining PRA results for significant contributors for the configuration, to aid in identifying appropriate compensatory measures (e.g., related to risk-significant systems that may provide diverse protection, or important support systems or human actions). Enclosure 9 identifies several areas of uncertainty in the internal events and fire PRAs that will be considered in defining configuration-specific RMAs when entering a RICT.

If the planned activity or emergent condition includes a SSC that is identified to impact Fire PRA, as identified in the current Configuration Risk Management Program (CRMP), Fire PRA specific RMAs associated with that SSC shall be implemented per the current plant procedure.

Risk Management Action Examples

Approved equipment-specific RMAs for risk significant SSCs within the scope of the RICT program will be contained in the procedure.

It is possible to credit RMAs in RICT calculations, to the extent the associated plant equipment and operator actions are modeled in the PRA; however, such quantification of RMAs is neither required nor expected by NEI 06-09. Nonetheless, if RMAs will be credited to determine RICTs, the procedure instructions will be consistent with the guidance in NEI 06-09.

NEI 06-09 classifies RMAs into the three categories described below:

1) Actions to increase risk awareness and control.

- Shift brief
- Pre-job brief
- Training
- Presence of system engineer or other expertise related to the activity
- Special purpose procedure to identify risk sources and contingency plans

2) Actions to reduce the duration of maintenance activities.

- Pre-staging materials
- Conducting training on mock-ups
- Performing the activity around the clock
- Performing walk-downs on the actual system(s) to be worked on prior to beginning work

3) Actions to minimize the magnitude of the risk increase.

- Suspend or minimize activities on redundant systems
- Suspend or minimize activities on other systems that adversely affect the CDF or LERF
- Suspend or minimize activities on systems that may cause a trip or transient to minimize the likelihood of an initiating event that the out-of-service component is meant to mitigate
- Use temporary equipment to provide backup power, ventilation, etc.
- Reschedule other risk-significant activities

4. Examples

Example RMAs that may be considered during a RICT Program entry for a Diesel Generator (DG) or a Containment Spray (CS) Pump to reduce the risk impact and ensure adequate defense-in-depth are:

Risk Management Action Examples

A. Diesel Generator:

1. Contact the Transmission System Operator (TSO) to determine the reliability of offsite power supplies prior to entering a RICT, and implement RMAs during times of high grid stress conditions, such as during high demand conditions.

2. Evaluate weather conditions for threats to the reliability of offsite power supplies.

3. Defer elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with the unit.

4. Defer planned maintenance or testing that affects the reliability of the operable DGs and their associated support equipment. Defer planned maintenance activities on station blackout mitigating systems. Treat these as protected equipment.

5. Defer planned maintenance or testing on redundant train safety systems. If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.

6. Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected DG.

7. Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review the appropriate emergency operating procedures for a Loss of Offsite Power.

B. Containment Spray Pump

1. Defer planned maintenance or testing activities on the redundant CS train and its associated support equipment, and treat those systems as protected equipment.

2. Defer planned maintenance or testing that affects the reliability of those safety systems that provide a defense-in-depth, such as Containment Air Coolers (CACs) or Emergency Core Cooling Systems (ECCS). If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.

3. Ensure all required materials, tools, and personnel are available, prior to entering the RICT, and perform maintenance activities around the clock.

4. Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established.

5. References

1. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322).

2. Exelon Procedure OP-AA-201-012-1001, "Operations On-Line Fire Risk Management."

3. Exelon Procedure OP-CA-201-012-1001, "On-Line Fire Risk Management."

4. Exelon Procedure WC-AA-101-1006, "On-Line Risk Management and Assessment."

5. Exelon Procedure OP-AA-108-117, "Protected Equipment Program."

6. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," U.S. Nuclear Regulatory Commission, March 2009.

7. EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," Technical Update, Electric Power Research Institute, December 2012.