FPL.

December 23, 2014
L-2014-369
10 CFR 50.90

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

Re: Turkey Point Nuclear Plant, Units 3 and 4
Docket Nos. 50-250 and 50-251
Renewed Facility Operating Licenses DRR-31 and DPR-41
License Amendment Request No. 236
Revision to the Technical Specifications to Adopt Risk Informed Completion Times
TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4B"

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), Florida Power & Light Company (FPL) is submitting a request for an amendment to the Technical Specifications (TS) for Turkey Point Nuclear Plant, Units 3 and 4.

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 change, the requested confirmation of applicability, and plant-specific verifications.
- Attachment 2 provides the existing TS pages marked up to show the proposed changes.
- Attachment 3 provides revised (clean) TS pages.
- Attachment 4 provides existing TS Bases pages marked up to show the proposed changes.

The proposed changes to the TS Bases are provided for information only and will be implemented in accordance with the TS Bases Control Program upon implementation of the requested amendment.

This license amendment request contains no new regulatory commitments and does not modify any existing commitments.

FPL requests approval of the proposed license amendment by January 31, 2016 with the amendment being implemented within 180 days.

A001
NRR

In accordance with 10 CFR 50.91(a)(1), "Notice for Public Comment," the analysis about the issue of no significant hazards consideration using the standards in 10 CFR 50.92 is being provided to the Commission.

This license amendment request has been reviewed by the Turkey Point Plant Nuclear Safety Committee.

In accordance with 10 CFR 50.91(b)(1), "Notice for Public Comment; State Consultation," a copy of this application with attachments is being provided to the designated Florida Official.

This license amendment request contains proposed changes to the same TS that are included in License Amendment Request (LAR) No. 212, "Proposed Changes to Turkey Point Technical Specifications (TS) Regarding Non-Conservative Action and Surveillance Requirement in TS 3/4.5.2," which FPL submitted on November 13, 2014. Following receipt of an amendment in response to LAR No. 212, FPL intends to supplement this LAR with proposed changes to the amended TS to incorporate risk informed completion times.

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 23, 2014

Sincerely,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachments

Enclosures

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Senior Resident Inspector, Turkey Point Nuclear Plant
Ms. Cindy Becker, Florida Department of Health

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ATTACHMENT 1

DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

License Amendment Request for Adoption of TSTF-505, Revision 1, "Provide Risk- Informed Extended Completion Times - RITSTF Initiative 4b"

1.0 DESCRIPTION

The proposed amendment would modify the Technical Specification (TS) requirements related to Completion Times (CTs) for Required Actions (RAs) 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 6.0, "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 Enclosure 1 are proposed to be changed, which does not include all of the modified Required Actions in TSTF-505 and which includes some plant-specific Required Actions not included in TSTF-505.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Florida Power & Light Company (FPL) has reviewed the model safety evaluation dated November 29, 2011 as part of the Federal Register Notice for Comment. 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, FPL 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 Turkey Point Nuclear Plant, Units 3 and 4, and support incorporation of this amendment in the Turkey Point TS.

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 CDF and 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 FPL'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

FPL is proposing optional changes and variations from the TS changes described in TSTF-505, Revision 1. Table 1 identifies each TS in TSTF-505 and the corresponding Turkey Point TS. The Table identifies and justifies any differences between the plant-specific TS and TSTF-505. The following is a discussion of the changes and variations.

1. The Turkey Point TS utilize different format, numbering, and titles than the Standard TS on which TSTF-505 was based. Also, the TS Bases do not contain the extent of information included in the Standard TS Bases regarding CTs. The Turkey Point TS Bases are revised where they contain discussions of CTs. In addition, revised Bases are provided to discuss CTs and the associated note for RAs that use a RICT for emergent conditions that result in a loss of a specified safety function or inoperability of all required trains of a system required to be operable (when one or more of the trains are considered PRA functional). These administrative differences in the TS are editorial in nature and do not affect the applicability of TSTF-505 to the Turkey Point TS.

2. FPL will implement RICTs in Modes 1 and 2 only and will not adopt changes in TSTF-505 for RAs that are applicable in Mode 3 and below. These cases are noted as optional changes in Table 1. NEI 06-09 states that PRAs that support RMTS are typically at-power PRAs directly applicable to plant configurations during operation in Modes 1 and 2.

3. FPL will not apply RICTs to functions that are not modeled in the PRA. These instances are noted as optional changes in Table 1. Consistent with NEI 06-09, the RMTS program defines the scope of equipment used to define plant configurations to which calculation of a RICT may be applied.

4. In some cases, the Turkey Point RAs require a shutdown and provide no restoration time, or the RA has a CT of 30 days or longer. TSTF-505 does not modify RAs with a CT of 30 days or more, or RAs that require a shutdown. A RICT will not be applied in these cases, which are noted as variations in Table 1.

5. Table 1 identifies as variations those instances where the Turkey Point TS do not contain a RA corresponding to a RA in TSTF-505.

6. TSTF-505 contains TS 3.3.5, Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation, with a Condition that addresses one or more functions with two or more channels per bus inoperable. However, the Turkey Point TS do not include a separate Limiting Condition for Operation (LCO) for LOP DG Start Instrumentation; the loss of voltage and degraded voltage functions are included in TS 3.3.2, ESFAS. Consistent with TSTF-505, FPL proposes to add new Action 18a to address the condition in which two channels per bus are inoperable.

7. TS 3.5.2, ECCS - Operating, in TSTF-505 requires two operable trains of ECCS. Condition A addresses one or more inoperable trains and assumes that with one or more ECCS trains inoperable, at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. TSTF-505 added new Condition B with a RICT to address a situation with less than 100% flow equivalent of a single ECCS train. The Actions in Turkey Point TS 3.5.2 do not specifically address ECCS flow capability, but instead address inoperability of ECCS components. (The ECCS flow assumed for a large break LOCA is provided by one residual heat removal pump and

two safety injection pumps.) Therefore, FPL proposes to apply the RICT Program to TS 3.5.2, Actions a, d, and e, for inoperable ECCS components.

FPL also proposes to provide new Actions h, i, j, k, and l that address combinations of inoperable ECCS components that result in less than the assumed minimum equipment. Each Action includes a note similar to TSTF-505 that prohibits use of the Action when the ECCS equipment is intentionally made inoperable. As discussed in TSTF-505, NEI 06-09 allows the application of a RICT to emergent conditions that represent a loss of a specified safety function or inoperability of all required trains or divisions of a system required to be operable provided one or more of the trains or divisions are considered "PRA functional." This proposed change adds RAs for configurations that currently would require entry into LCO 3.0.3. The proposed variations are consistent with the changes in TSTF-505, while they accommodate the differences in content of TSTF-505 and the Turkey Point TS.

8. The Turkey Point TS contain several plant-specific Actions (not included in TSTF-505) that specify a CT for restoring the capability of powering operable equipment from the associated operable emergency diesel generator (EDG). FPL proposes to apply a RICT to these actions. TSTF-505 applies a RICT to the RA for restoration of an EDG to operable status, and FPL proposes to adopt this change. Therefore, application of a RICT to the RAs to restore the capability to power operable equipment from the associated EDG is appropriate and aligns the CTs with that for an inoperable EDG. This is a necessary conforming change for adopting a RICT in TS 3.8.1.1, A.C. Sources - Operating.

9. TSTF-505 contains one omission (editorial error) that is noted as an editorial change in Table 1. TS 3.3.1, Reactor Trip System Instrumentation, Function 20 in Table 3.3-1 of TSTF-505 only identifies the Condition associated with one channel inoperable. New Condition JJ in TS 3.3.1 applies with one trip mechanism inoperable for two or more reactor trip breakers. However, Condition JJ is omitted from Table 3.3-1. FPL proposes to apply this Condition and a RICT to the corresponding Turkey Point TS (Table 3.3-1, Functional Unit 19, Action 10, 10A).

10. TS 3.7.2 for main steam isolation valves (MSIVs) in TSTF-505 provides for unlimited operation in Mode 2 with one or more inoperable MSIVs provided the inoperable MSIVs are closed. The Condition applicable in Mode 1 for two or more inoperable MISVs (Condition C) requires a shutdown to Mode 4 if the MSIVs are not restored to operable. This structure is not correct because once Mode 1 is exited, continued operation in Mode 2 or 3 is permitted with MSIVs inoperable but closed. In order to establish the correct structure for Turkey Point TS 3.7.1.5, FPL proposes that the RAs for one or more inoperable MSIVs in Mode 1 require entering Mode 2, rather than Modes 3 and 4, if the MSIVs are not restored to operable status within the CT. This change is not a plant-specific variation but rather addresses a previously identified issue with the structure of TS 3.7.2 in TSTF-505.

In addition, FPL proposes to revise the RA in Turkey Point TS 3.7.1.5 that is applicable for an inoperable MSIV in Modes 2 and 3 to address one or more inoperable MSIVs. This change is a variation because Condition D in TSTF-505 already addresses one or more inoperable MSIVs. This change aligns the Turkey Point TS with TSTF-505 and is appropriate because it provides a subsequent RA if entry into Mode 2 is required because two or more inoperable MSIVs in Mode 1 are not restored to operable status within the CT.

11. An oversight occurred during the NRC review of TSTF-505, Revision 1, and the "-A" was omitted from the reference to NEI 06-09, Revision 0, in the Risk-Informed Completion Time Program in proposed TS Section 5.5.18. FPL has revised corresponding Turkey Point proposed TS 6.8.4.l, Risk Informed Completion Time Program, with the reference as follows: NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b: Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, November 2006.

12. The proposed changes include corrections of misspellings in the TS. These editorial changes are administrative in nature and do not affect the model safety evaluation or the applicability of TSTF-505 to the Turkey Point TS.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.3.1 Function 1, manual reactor trip Condition B, C	Table 3.3-1, Functional Unit 1 Actions 1 and 1A	Yes	
3.3.1 Function 1, manual reactor trip Condition D, E	Table 3.3-1, Functional Unit 1	No	OPTIONAL CHANGE - This function is applicable in Modes 3 - 5. FPL is not proposing to apply the RICT Program in Modes 3 - 5.
3.3.1 Function 2.a, power range neutron flux - High Condition F, G	Table 3.3-1, Functional Unit 2.a	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 2.b, power range neutron flux - Low Condition H, I	Table 3.3-1, Functional Unit 2.b	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 3.a, power range neutron flux rate - high positive rate Condition H, I	N/A	N/A	VARIATION - Turkey Point TS do not contain this function.
3.3.1 Function 3.b, power range neutron flux rate - high negative rate Condition H, I	N/A	N/A	VARIATION - Turkey Point TS do not contain this function.
3.3.1 Function 4, Intermediate range neutron flux Condition J, K	Table 3.3-1, Functional Unit 3	N/A	EDITORIAL - TSTF-505 only renumbered the conditions. This change is not applicable to the Turkey Point TS.
3.3.1 Function 5, source range neutron flux Condition L, M	Table 3.3-1, Functional Unit 4	N/A	EDITORIAL - TSTF-505 only renumbered the conditions. This change is not applicable to the Turkey Point TS.
3.3.1 Function 6, overtemperature ΔT Condition H	Table 3.3-1, Functional Unit 5	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 7, overpower ΔT Condition H	Table 3.3-1, Functional Unit 6	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.

TABLE 1

Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.3.1 Function 8.a, pressurizer pressure - Low Condition P	Table 3.3-1, Functional Unit 7	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 8.b, pressurizer pressure - High Condition H	Table 3.3-1, Functional Unit 8	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 9, pressurizer water level - High Condition P	Table 3.3-1, Functional Unit 9	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 10, reactor coolant flow - Low Condition P	Table 3.3-1, Functional Unit 10	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 11, RCP breaker position Condition S, T, V, W	Table 3.3-1, Functional Unit 18	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 12, undervoltage RCPs Condition P	Table 3.3-1, Functional Unit 13	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 13, underfrequency RCPs Condition P	Table 3.3-1, Functional Unit 14	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 14, SG water level low Condition H	Table 3.3-1, Functional Unit 11	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 15, SG water level low with steam flow/feed flow mismatch Condition H	Table 3.3-1, Functional Unit 12	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 16.a, turbine trip Condition Y, Z	Table 3.3-1, Functional Unit 15	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.

TABLE 1

Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.3.1 Function 17, SI input from ESFAS Condition BB	Table 3.3-1, Functional Unit 16	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.1 Function 18, reactor trip system interlocks Condition FF, GG	Table 3.3-1, Functional Unit 17	N/A	EDITORIAL - TSTF-505 made editorial changes only.
3.3.1 Function 19, reactor trip breakers Condition DD, EE	Table 3.3-1, Functional Unit 19, Action 8	No	VARIATION - The Turkey Point TS Action does not provide a restoration time; therefore, FPL is not proposing to apply the RICT Program to this Action.
3.3.1 Function 19, reactor trip breakers Condition D	Table 3.3-1, Functional Unit 19, Action 9	No	OPTIONAL CHANGE - This function is applicable in Modes 3 - 5. FPL is not proposing to apply the RICT Program in Modes 3 - 5.
3.3.1 Function 20, reactor trip breakers undervoltage and shunt trip mechanisms Condition II	Table 3.3-1, Functional Unit 19, Action 10, 10A	Yes	EDITORIAL - TSTF-505, Condition JJ for one trip mechanism inoperable on two or more reactor trip breakers should apply to this function.
3.3.1 Function 21, Automatic Trip Logic Condition BB	Table 3.3-1, Functional Unit 19, Action 8	No	VARIATION - The Turkey Point TS Action does not provide a restoration time; therefore, FPL is not proposing to apply the RICT Program to this Action.
3.3.1 Function 21, Automatic Trip Logic Condition D	Table 3.3-1, Functional Unit 20, Action 9	No	OPTIONAL CHANGE - This function is applicable in Modes 3 - 5. FPL is not proposing to apply the RICT Program in Modes 3 - 5.
3.3.2 Function 1.a, SI manual initiation Condition B, C	Table 3.3-2, Functional Unit 1.a, Action 28, 29	Yes	

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.3.2 Function 1.b, auto actuation and logic actuation relays Condition D, E	Table 3.3-2, Functional Unit 1.b	No	VARIATION - The Turkey Point TS Action does not provide a restoration time; therefore, FPL is not proposing to apply the RICT Program to this Action.
3.3.2 Function 1.c, containment pressure - High 1 Condition F, G	Table 3.3-2, Functional Unit 1.c Action 26, 27	Yes	
3.3.2 Function 1.d, pressurizer pressure - Low Condition F, G	Table 3.3-2, Functional Unit 1.d Action 26, 27	Yes	
3.3.2 Function 1.e.(1), steam line pressure - Low Condition F, G	N/A	N/A	VARIATION - Turkey Point TS do not contain this function.
3.3.2 Function 1.e.(2), high differential pressure between steam lines Condition F, G	Table 3.3-2, Functional Unit 1.e Action 26, 27	Yes	
3.3.2 Function 1.f, high steam flow in two steam lines with Tavg low-low Condition F, G	Table 3.3-2, Functional Unit 1.f Action 25	Yes	
3.3.2 Function 1.g, high steam flow in two steam lines with steam line pressure low Condition F, G	Table 3.3-2, Functional Unit 1.f Action 26, 27	Yes	
3.3.2 Function 2.a, containment spray manual initiation Condition B, C	N/A	No	VARIATION - Turkey Point TS do not contain this function.
3.3.2 Function 2.b, containment spray auto actuation logic and actuation relays Condition D, E	Table 3.3-2, Functional Unit 2.a	No	VARIATION - The Turkey Point TS Action does not provide a restoration time; therefore, FPL is not proposing to apply the RICT Program to this Action.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.3.2 Function 2.c, containment spray containment pressure high Condition H, I	Table 3.3-2, Functional Unit 2.b Action 26, 27	Yes	VARIATION - TS 3.3.2, Condition H in TSTF-505 for an inoperable containment pressure channel does not apply a RICT because the RA is to place the channel in bypass. The Turkey Point TS require placing the inoperable channel in the tripped condition; therefore, consistent with TSTF-505, FPL proposes to apply a RICT for placing the channel in the tripped condition.
3.3.2 Function 3.a.(1), containment phase A isolation - manual initiation Condition B, C	Table 3.3.2, Functional Unit 3.a.(1) Action 28, 29	Yes	
3.3.2 Function 3.a.(2), containment phase A isolation auto actuation logic and actuation relays Condition D, E	Table 3.3.2, Functional Unit 3.a.(2)	No	VARIATION - The Turkey Point TS Action does not provide a restoration time; therefore, FPL is not proposing to apply the RICT Program to this Action.
3.3.2 Function 3.b.(1), containment phase B isolation - manual initiation Condition B, C	Table 3.3.2, Functional Unit 3.b.(1) Action 28, 29	Yes	
3.3.2 Function 3.b.(2), containment phase B isolation auto actuation logic and actuation relays Condition D, E	Table 3.3.2, Functional Unit 3.b.(2)	No	VARIATION - The Turkey Point TS Action does not provide a restoration time; therefore, FPL is not proposing to apply the RICT Program to this Action.
3.3.2 Function 3.b.(3), containment phase B isolation - containment pressure high 3 Condition H, I	Table 3.3.2, Functional Unit 3.b.(3)	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.2 Function 4, steam line isolation Condition J, K, L, M, F, G	Table 3.3.2, Functional Unit 4	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.

TABLE 1

Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.3.2 Function 5, turbine trip and feedwater isolation Condition N, O, L, M, P, Q, F, G	Table 3.3.2, Functional Unit 5	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.2 Function 6.a, auxiliary feedwater automatic actuation logic and actuation relays Condition L, M	Table 3.3.2, , Functional Unit 6.a	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.2 Function 6.b, auxiliary feedwater automatic actuation logic and actuation relays Condition L, M	N/A	N/A	VARIATION - Turkey Point TS do not contain this function.
3.3.2 Function 6.c, auxiliary feedwater SG water level - low low Condition F, G	Table 3.3.2, Functional Unit 6.b Action 26, 27	Yes	
3.3.2 Function 6.e, auxiliary feedwater loss of offsite power Condition J, K	Table 3.3.2, , Functional Unit 6.d	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.2 Function 6.f, auxiliary feedwater undervoltage RCP Condition P, Q	N/A	N/A	VARIATION - Turkey Point TS do not contain this function.
3.3.2 Function 6.g, auxiliary feedwater trip of all main feedwater pumps Condition R, S	Table 3.3.2, Functional Unit 6.e	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.2 Function 7, Auto switchover to containment sump Condition D, E, T, U	N/A	N/A	VARIATION - Turkey Point TS do not contain this function.
3.3.2 Function 8.a, ESFAS interlocks P-4 Condition J, K	N/A	N/A	VARIATION - Turkey Point TS do not contain this function.

TABLE 1

Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.3.2 Function 8.b, ESFAS interlocks P-11 Condition V	Table 3.3.2, Functional Unit 8.a	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.2 Function 8.a, ESFAS interlocks P-12 Condition V	Table 3.3.2, Functional Unit 8.b	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.3.5 A.1 LOP DG Start Instrumentation Condition A	Table 3.3.2, ESFAS Functional Unit 7.a, loss of power 4.16 kV buses A and B - Action 18 Functional Unit 7.b, loss of power 480 V load centers - undervoltage Action 18 Functional Unit 7.c, loss of power 480 V load centers - degraded voltage Action 18	Yes	
3.3.5 B.1 LOP DG Start Instrumentation	TS 3.3.2, Action 18a	Yes	VARIATION - The Turkey Point TS do not currently include an action to address two channels inoperable on any bus as provided for in the Standard TS and TSTF-505. FPL proposes to adopt a new action with a one-hour CT and the option to use the RICT Program, consistent with TSTF-505.
3.3.9 BDPS	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.4.5 A.1 RCS Loops - Mode 3	TS 3.4.1.2	No	OPTIONAL CHANGE - This RA is applicable in Mode 3. FPL is not proposing to apply the RICT Program in Mode 3.
3.4.5 C.1 RCS Loops - Mode 3	TS 3.4.1.2	No	OPTIONAL CHANGE - This RA is applicable in Mode 3. FPL is not proposing to apply the RICT Program in Mode 3.
3.4.5 C.2 RCS Loops - Mode 3	TS 3.4.1.2	No	OPTIONAL CHANGE - This RA is applicable in Mode 3. FPL is not proposing to apply the RICT Program in Mode 3.
3.4.9 B.1 Pressurizer	TS 3.4.3	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.4.9 C.1 Pressurizer	TS 3.4.3	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.4.10 A.1 Pressurizer Safety Valves	TS 3.4.2.2	Yes	
3.4.11 B.3 Pressurizer PORVs	TS 3.4.4, Action b	No	VARIATION - The Turkey Point TS permit continued operation with one PORV inoperable with its associated block valve closed and power removed from the block valve. Therefore, FPL is not proposing to apply the RICT Program to this Action.
3.4.11 C.2 Pressurizer PORVs	TS 3.4.4, Action d	No	VARIATION - The Turkey Point TS provide 30 days to restore an inoperable block valve when both are inoperable. Therefore, FPL is not proposing to apply the RICT Program to this Action.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.4.11 E.3 Pressurizer PORVs	TS 3.4.4, Action c	No	VARIATION - The Turkey Point TS provide 30 days to restore an inoperable PORV when both are inoperable. Therefore, FPL is not proposing to apply the RICT Program to this Action.
3.4.11 F.1 Pressurizer PORVs	TS 3.4.4, Action d	No	VARIATION - The Turkey Point TS provide 30 days to restore an inoperable block valve when both are inoperable. Therefore, FPL is not proposing to apply the RICT Program to this Action.
3.4.14 C.1 RCS PIV Leakage	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.5.1 A.1 Accumulators	TS 3.5.1, Action b	Yes	
3.5.1 B.1 Accumulators	TS 3.5.1, Action a	Yes	
3.5.1 C.1 Accumulators	TS 3.5.1, Action c	Yes	

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.5.2 A.1 ECCS - Operating	TS 3.5.2 - Action a TS 3.5.2 - Action d TS 3.5.2 - Action e TS 3.5.2 - Action g	Yes	VARIATION - TS 3.5.2, Condition A in TSTF-505 assumes that with one or more ECCS trains inoperable, at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. The Actions in Turkey Point TS 3.5.2 do not specifically address ECCS flow capability. The Actions address inoperability of ECCS components, but operation in accordance with these actions maintains the minimum assumed ECCS flow capability, which is provided by one residual heat removal (RHR) pump and two safety injection pumps. Therefore, FPL proposes to apply the RICT Program to TS 3.5.2, Actions a, d, and e, which are consistent with the intent of TS 3.5.2, Condition A in TSTF-505.
N/A	TS 3.5.2 - Action f	Yes	VARIATION - Turkey Point TS 3.5.2, Action f addresses the condition of an operable safety injection pump not capable of being powered from its associated emergency diesel generator (EDG). TSTF-505 does not contain a similar Condition and RA. The time provided in Action f to restore this capability is the same as the time provided in TS 3.8.1.1 to restore an inoperable EDG to operable status. FPL proposes to apply the RICT to TS 3.5.2, Action f. This is a necessary conforming change to permit adopting a RICT in TS 3.8.1.1, A.C. Sources - Operating.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.5.2 B.1 ECCS - Operating	TS 3.5.2 - Action h TS 3.5.2 - Action i TS 3.5.2 - Action j TS 3.5.2 - Action k TS 3.5.2 - Action l	Yes	VARIATION - TS 3.5.2, Condition A in TSTF-505 addresses inoperable ECCS trains where at least 100% of the ECCS flow equivalent to a single operable ECCS train is available and Condition B addresses inoperable ECCS trains with less than 100% of the ECCS flow equivalent available. The Actions in Turkey Point TS 3.5.2 do not specifically address ECCS flow capability. The minimum ECCS components required to provide the assumed flow are one RHR pump and two safety injection pumps. FPL proposes new Actions h, i, j, k, and l that address combinations of inoperable ECCS components that result in less than the assumed minimum equipment. The proposed change is consistent with the intent of TS 3.5.2, Condition B.1 in TSTF-505. Each Action includes a note similar to TSTF-505 that prohibits use of the Action when the ECCS equipment is intentionally made inoperable.
3.5.3 B.1 ECCS - Shutdown	TS 3.5.3	No	OPTIONAL CHANGE - This RA is applicable in Mode 4. FPL is not proposing to apply the RICT Program in Mode 4.
3.5.4 A.1 RWST	TS 3.5.4	Yes	VARIATION - Turkey Point TS specifies the number of required refueling water storage tanks (RWSTs) and has only a single Action for an inoperable RWST, which does not distinguish the cause of the inoperability. FPL proposes to apply the RICT Program to the single Action for an inoperable RWST.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.5.4 B.1 RWST	TS 3.5.4	Yes	VARIATION - Turkey Point TS specifies the number of required RWSTs and has only a single Action for an inoperable RWST, which does not distinguish the cause of the inoperability. FPL proposes to apply the RICT Program to the single Action for an inoperable RWST.
3.5.6 A.1 BIT	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.
3.6.2 C.3 Containment Air Locks	TS 3.6.1.3, Action b	Yes	
3.6.3 A.1 Containment Isolation valves	TS 3.6.4, Actions a, b, c	Yes	VARIATION - Turkey Point TS Actions are structured differently than the RAs in TSTF-505 and do not distinguish the type of penetration. Actions b and c each isolate the penetration using a different method. TSTF-505 includes all of the acceptable methods for isolating a penetration in a single RA for each type of penetration in RAs A.1, B.1, and C.1. Turkey Point TS Action a is a "restore to OPERABLE status" action, and TSTF-505 does not contain a similar action; restoration to operability is always an option whether stated or not. FPL proposes to apply the RICT Program to TS 3.6.4, Actions a, b, and c.
3.6.3 A.2 Containment Isolation valves	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.6.3 B.1 Containment Isolation valves	TS 3.6.4, Actions a, b, c	Yes	See notes above for 3.6.3.A.1.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.6.3 C.1 Containment Isolation valves	TS 3.6.4, Actions a, b, c	Yes	See notes above for 3.6.3.A.1.
3.6.3 C.2 Containment Isolation valves	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.6.3 D.1 Containment Isolation valves	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.6.3 E.1 Containment Isolation valves	TS 3.6.1.7, Action a TS 3.6.1.7, Action b	Yes	The Turkey Point TS contain a separate LCO for the containment purge isolation valves where these valves are included in TS 3.6.3 in TSTF-505. Application of the RICT Program to TS 3.6.1.7, Actions a and b is consistent with the treatment of these valves in TSTF-505 under TS 3.6.3, RA E.1.
3.6.3 E.2 Containment Isolation valves	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.6.3 E.3 Containment Isolation valves	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.6.6B A.1 Containment Spray and Cooling Systems	TS 3.6.2.1, Action a	Yes	
3.6.6B B.1 Containment Spray and Cooling Systems	TS 3.6.2.2, Action a	Yes	
3.6.6B C.1 Containment Spray and Cooling Systems	TS 3.6.2.1, Action b	Yes	
3.6.6B D.1 Containment Spray and Cooling Systems	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.6.6B D.2 Containment Spray and Cooling Systems	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.6.6B E.1 Containment Spray and Cooling Systems	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.6.6B G.1 Containment Spray and Cooling Systems	TS 3.6.2.2, Action b	Yes	VARIATION - Turkey Point TS differ from TSTF-505 and contain separate LCOs for containment spray and containment cooling systems. Turkey Point TS 3.6.2.2, Action b addresses two or more inoperable emergency containment cooling units with a one hour CT to restore two cooling units.
3.6.9 B.2 HMS	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.
3.6.10 A.1 B.1 C.1 HIS	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.
3.6.14 A.1 B.1 ARS	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.
3.6.15 A.1 Ice Bed	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.
3.6.16 A.1 B.2 Ice Condenser Doors	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.
3.6.17 A.1 B.1 Divider barrier Integrity	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.6.18 A.1 B.1 Containment Recirculation Drains	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.
3.7.2 A.1 MSIVs	TS 3.7.1.5, Action a	Yes	FPL proposes that the RA for one inoperable MSIV in Mode 1 requires entering Mode 2, rather than Modes 3 and 4, if the MSIV is not restored to operable status within the CT. This change is not a plant-specific variation but rather addresses a previously identified issue with the structure of TS 3.7.2 in TSTF-505.
3.7.2 C.1 MSIVs	TS 3.7.1.5, Action b	Yes	FPL proposes that the RA for two or more inoperable MSIVs in Mode 1 require entering Mode 2, rather than Modes 3 and 4, if the MSIVs are not restored to operable status within the CT. This change is not a plant-specific variation but rather addresses a previously identified issue with the structure of TS 3.7.2 in TSTF-505.
N/A	TS 3.7.1.5 Action in Modes 2 and 3	N/A	FPL proposes to revise the RA in Turkey Point TS 3.7.1.5 that is applicable for an inoperable MSIV in Modes 2 and 3 to address one or more inoperable MSIVs. This change is a variation because Condition D in TSTF-505 already addresses one or more inoperable MSIVs. This change aligns the Turkey Point TS with TSTF-505 and is appropriate because it provides a subsequent RA if entry into Mode 2 is required because two or more inoperable MSIVs in Mode 1 are not restored to operable status within the CT.
3.7.4 ADVs	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.7.5 A.1 AFW	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.7.5 B.1 AFW	TS 3.7.1.2, Action 1	Yes	
3.7.5 C.1 AFW	N/A	N/A	VARIATION - Turkey Point TS do not contain this TS.
N/A	TS 3.7.1.2, Action 2	Yes	VARIATION - Turkey Point TS 3.7.1.2 requires two operable AFW trains and three AFW pumps. Current TS Action 2 addresses two inoperable AFW trains and provides a two-hour Completion Time to restore at least one train to operable status. FPL proposes to apply the RICT to TS 3.7.1.2, Action 2. VARIATION - Current Turkey Point TS 3.7.1.2, Action 2, addresses inoperability of both auxiliary feedwater trains. FPL proposes to modify the Action with a note that states the RICT Program is not applicable when the AFW trains are intentionally made inoperable. This is consistent with NEI 06-09, which states that voluntary use of the RMTS for a configuration which represents a loss of TS specified safety function, or inoperability of all required safety trains, is not permitted. However, consistent with the current licensing bases, the Action and front stop CT are applicable to a voluntary entry into the condition.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.7.6 A.2 CST	TS 3.7.1.3	Yes	VARIATION - Turkey Point TS 3.7.1.3 contains three different actions for an inoperable CST depending on the Mode of operation of the opposite unit and the available CST volume. FPL proposes to apply the RICT Program to the three Actions in TS 3.7.1.3.
3.7.7 A.1 CCW	TS 3.7.2, Action b	Yes	VARIATION - Condition 3.7.7, A.1 addresses an inoperable train while the corresponding Turkey Point TS addresses inoperability on a component level (pumps and heat exchangers). This variation is the result of different structure of the Turkey Point TS; however, the proposed change is consistent with the intent of TSTF-505.
3.7.7 B.1 CCW	TS 3.7.2, Action c, d	Yes	VARIATION - Condition 3.7.7, B.1 addresses inoperable trains while the corresponding Turkey Point TS addresses inoperability on a component level (pumps and heat exchangers). This variation is the result of different structure of the Turkey Point TS; however, the proposed change is consistent with the intent of TSTF-505.
3.7.8 A.1 SWS	TS 3.7.3, Action a, b, c	Yes	VARIATION - Condition 3.7.8, A.1 addresses an inoperable train while the corresponding Turkey Point TS addresses inoperability on a component level (pumps and headers). This variation is the result of different structure of the Turkey Point TS; however, the proposed change is consistent with the intent of TSTF-505.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.7.8 B.1 SWS	TS 3.7.3, Action d, e	Yes	VARIATION - Condition 3.7.8, B.1 addresses inoperable trains while the corresponding Turkey Point TS addresses inoperability on a component level (pumps and headers). This variation is the result of different structure of the Turkey Point TS; however, the proposed change is consistent with the intent of TSTF-505.
3.7.9 A.1 UHS	TS 3.7.4	No	VARIATION - The Turkey Point TS Action does not provide a restoration time; therefore, FPL is not proposing to apply the RICT Program to this Action.
3.7.9 C.1 UHS	TS 3.7.4	No	VARIATION - The Turkey Point TS Action does not provide a restoration time; therefore, FPL is not proposing to apply the RICT Program to this Action.
3.7.11 B.1 CREATCS	TS 3.7.5	No	OPTIONAL CHANGE - This function is not modeled in the PRA. FPL is not proposing to apply the RICT Program to this function.
3.8.1. A.3 AC Sources - Operating	TS 3.8.1.1, Action a	Yes	
3.8.1. B.4 AC Sources - Operating	TS 3.8.1.1, Action b	Yes	
3.8.1. C.2 AC Sources - Operating	TS 3.8.1.1, Action e	Yes	
3.8.1. D.1 AC Sources - Operating	TS 3.8.1.1, Action c TS 3.8.1.1, Action a	Yes	VARIATON - Turkey Point TS 3.8.1.1, Action c addresses concurrent inoperability of one AC source and one EDG. However, Action c does not provide a restoration time but instead directs restoration of an inoperable AC source in accordance with Action a.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.8.1. D.2 AC Sources - Operating	TS 3.8.1.1, Action c TS 3.8.1.1, Action b	Yes	VARIATION - Turkey Point TS 3.8.1.1, Action c addresses concurrent inoperability of one AC source and one EDG. However, Action c does not provide a restoration time but instead directs restoration of an inoperable EDG in accordance with Action b.
3.8.1. E.1 AC Sources - Operating	TS 3.8.1.1, Action f	Yes	
3.8.1. F.1 AC Sources - Operating	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.8.1. G.1 AC Sources - Operating	TS 3.8.1.1, Action i	Yes	
3.8.4 A.1 DC Sources - Operating	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
N/A	TS 3.8.2.1, Action a	Yes	VARIATION - Turkey Point TS 3.8.2.1, Action a provides a restoration time when one or more of the required battery chargers is operable but not capable of being powered from its associated operable EDG. TSTF-505 does not contain a similar RA. FPL proposes to apply the RICT Program to Action a. This is consistent with TSTF-505, TS 3.8.4, RA A.3; and TS 3.8.1, RA B.4, which apply the RICT Program to restoration of an inoperable battery charger and restoration of an inoperable EDG, respectively.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.8.4 A.3 DC Sources - Operating	TS 3.8.2.1, Action b	Yes	VARIATION - Turkey Point Action b addresses both an inoperable battery bank and inoperable battery chargers associated with a battery bank.
3.8.4 B.1 DC Sources - Operating	TS 3.8.2.1, Action b	Yes	VARIATION - Turkey Point Action b addresses both an inoperable battery bank and inoperable battery chargers associated with a battery bank.
3.8.4 C.1 DC Sources - Operating	N/A	N/A	VARIATION - Turkey Point TS do not contain this RA.
3.8.4 D.1 DC Sources - Operating	TS 3.8.2.1, Action c	Yes	VARIATION - LCO 3.8.4 in TSTF-505 requires operability of "DC electrical power subsystems", while Turkey Point LCO 3.8.2.1 specifies operability requirements on a component level (battery banks and battery chargers). FPL proposes to alter the wording of new Action c from that of RA 3.8.4.D.1 in TSTF-505. Consistent with the intent of TSTF-505, this variation maintains consistency with the current TS by addressing inoperability on a component level.
3.8.7 A.1 Inverters - Operating	TS 3.8.3.1, Action c	Yes	The Turkey Point TS do not include a separate TS for Inverters. TS 3.8.3.1 establishes the requirements for inverters.
3.8.7 B.1 Inverters - Operating	TS 3.8.3.1, Action d	Yes	The Turkey Point TS do not include a separate TS for Inverters. TS 3.8.3.1 establishes the requirements for inverters.

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.8.9 A.1 Distribution Systems - Operating	TS 3.8.3.1, Action a	Yes	
N/A	TS 3.8.3.1, Action b	Yes	VARIATION - The LCO for Turkey Point TS 3.8.3.1 lists the specific electrical buses required to be operable where LCO 3.8.9 in TSTF-505 requires operable distribution subsystems but does not list all the required components. As a result of this difference, TS 3.8.3.1 contains Action b, which is not included in TSTF-505, to provide restoration times for inoperable load centers and motor control centers. FPL proposes to apply the RICT Program to the restoration times contained in Tables 3.8-1 and 3.8-2 associated with Action b. This variation is consistent with TS 3.8.9, Condition A.1, which applies the RICT Program when one or more AC electrical power distribution subsystems are inoperable.
3.8.9 B.1 Distribution Systems - Operating	TS 3.8.3.1, Action c TS 3.8.3.1, Action d	Yes	Turkey Point TS 3.8.3.1, Action c addresses the condition of one AC vital bus inoperable while RA 3.8.9 B.1 in TSTF-505 addresses one or more inoperable panels. New TS 3.8.3.1, Action d addresses the condition involving more than one inoperable AC panel and more than one inoperable inverter. This Action aligns with TS 3.8.9, Condition C.1, and TS 3.8.7, Condition B.1 in TSTF-505.
3.8.9 C.1 Distribution Systems - Operating	TS 3.8.3.1, Action e	Yes	

TABLE 1
Cross Reference of Required Actions (RA) in TSTF-505 and Corresponding RAs in Turkey Point TS

TSTF-505 TS Number and Required Action (RA)	Turkey Point Corresponding TS and RA	Apply RICT?	Notes
3.8.9 D.1 Distribution Systems - Operating	N/A	N/A	VARIATION - This new condition in TSTF-505 specifies a loss of safety function, which it is not permitted. Therefore, FPL does not propose to adopt this change.

3.0 REGULATORY SAFETY ANALYSIS

3.1 No Significant Hazards Consideration Determination

Florida Power & Light (FPL) has evaluated the proposed change to the TS using the criteria in 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration.

Turkey Point Nuclear Plant, Units 3 and 4 request 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 Section 6.0, "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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change permits 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 change does not involve a significant increase in the probability of an accident previously evaluated because the change involves no change to the plant or its modes of operation. The proposed change does 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.

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

   Response: No.

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

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

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

The proposed change permits the extension of Completion Times provided risk is assessed and managed in accordance with the NRC approved Risk-Informed Completion Time Program. The proposed change implements 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 change does not involve a significant reduction in a margin of safety.

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

4.0 ENVIRONMENTAL CONSIDERATION

FPL 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. FPL has concluded that the NRC staff findings presented in that evaluation are applicable to Turkey Point Nuclear Plant, Units 3 and 4.

The proposed change 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 change does 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 change meets 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 change.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UPS)

INSERT 1

or in accordance with the Risk Informed Completion Time Program,

INSERT 2

----------Note----------

Action h is not applicable when two or more of the required ECCS components or flow paths intentionally made inoperable.

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h. With two or more of the required ECCS components or flow paths inoperable, except for Safety Injection pumps and RHR pumps:

   1. Restore at least all but one of the inoperable components or flow paths to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

   2. With all but one inoperable ECCS component or flow path restored to OPERABLE status, comply with ACTION a for the remaining inoperable ECCS component or flow path.

----------Note----------

1. Action i is not applicable when three or more of the required Safety Injection pumps intentionally made inoperable.
2. Action i applies to both units simultaneously.

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i. With three or more of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3:

   1. Restore at least all but two inoperable Safety Injection pumps to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

   2. With all but two inoperable Safety Injection pumps restored to OPERABLE status, comply with ACTION d for the two remaining inoperable Safety Injection pumps.

----------Note----------

Action j is not applicable when two or more of the required Safety Injection pumps intentionally made inoperable.

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j. With two or more of the three required Safety Injection pumps inoperable and the opposite unit in MODE 4, 5, or 6:

   1. Restore at least all but one inoperable Safety Injection pump to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
   2. With all but one inoperable Safety Injection pump restored to OPERABLE status, comply with ACTION e for the remaining inoperable Safety Injection pump.

----------Note----------

Action k is not applicable when two or more required Safety Injection pumps are intentionally not powered from their associated diesel generator.

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k. With two or more required Safety Injection pumps OPERABLE but not capable of being powered from their associated diesel generator:

   1. Restore the capability of being powered from the associated diesel generator for at least all but one Safety Injection pump within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
   2. With all but one Safety Injection pump capable of being powered from the associated diesel generator, comply with ACTION f for the remaining Safety Injection pump not capable of being powered from its associated diesel generator.

----------Note----------

Action l is not applicable when two RHR pumps intentionally made inoperable.

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l. With two RHR pumps inoperable:

   1. Restore at least one inoperable RHR pump to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
   2. With one inoperable RHR pump restored to OPERABLE status, comply with ACTION g for the remaining inoperable RHR pump.

INSERT 3

----------Note----------

Action d is not applicable when ICW pumps intentionally made inoperable.

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d. With three ICW pumps inoperable:

   1. Restore at least one inoperable ICW pump to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   2. With one inoperable ICW pump restored to OPERABLE status, comply with ACTION b for the two remaining inoperable ICW pumps.

----------Note----------

Action e is not applicable when ICW headers intentionally made inoperable.

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e. With two ICW headers inoperable:

   1. Restore at least one inoperable ICW header to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   2. With one inoperable ICW header restored to OPERABLE status, comply with ACTION c for the remaining inoperable ICW header.

INSERT 4

----------Notes----------

1. Action c is not applicable when two or more battery banks or associated full capacity chargers intentionally made inoperable.
2. Action c applies to both units simultaneously.

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c. With two or more battery banks inoperable or with the full capacity chargers associated with two or more battery banks inoperable:

   1. Restore at least all but one inoperable battery bank to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours, and

   2. Restore one full-capacity battery charger associated with at least all but one battery bank to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours

3. With all but one inoperable battery bank restored to OPERABLE status, comply with ACTION b for the inoperable battery bank.
4. With all but one battery bank having at least one of its associated full-capacity battery chargers restored to OPERABLE status, comply with ACTION b for the inoperability of all battery chargers associated with the battery bank.

INSERT 5

I. 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, "Risk-Informed Technical Specifications Initiative 4b: Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, November 2006. The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 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.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1 , 1A
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	13
6. Overpower ΔT	3	2	2	1, 2	13
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	13
10. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Safety Injection Input from ESF	2	1	2	1, 2	8
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2#	7
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	7
or					
Turbine Inlet Pressure	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
18. Reactor Coolant Pump Breaker Position Trip					
a. Above P-8	1/breaker	1	1/breaker	1	11
b. Above P-7 and below P-8	1/breaker	2	1/breaker	1	11
19. Reactor Trip Breakers	2	1	2	1, 2	8, 10, 10A
	2	1	2	3*, 4*, 5*	9
20. Automatic Trip and Interlock logic	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

* When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

** When the Reactor Trip System breakers are in the open position, one or both of the backup NIS instrumentation channels may be used to satisfy this requirement. For backup NIS testing requirements, see Specification 3/4.3.3.3, ACCIDENT MONITORING.

*** Reactor Coolant Pump breaker A is tripped by underfrequency sensor UF-3A1(UF-4A1) or UF-3B1(UF-4B1). Reactor Coolant Pump breakers B and C are tripped by underfrequency sensor UF-3A2(UF-4A2) or UF-3B2(UF-4B2).

\# Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

\#\# Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

Insert 1

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored per Specification 4.2.4.2.

Note: Action 1A is not applicable when second Manual Reactor Trip channel intentionally made inoperable.

ACTION 1A - With two Manual Reactor Trip channels inoperable, restore at least one channel to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours. With one channel restored to OPERABLE status, comply with ACTION 1 for the remaining inoperable channel.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

Insert 1

ACTION 10- With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable, except for the time required for performing maintenance to restore the breaker to OPERABLE status.

Note: Action 10A is not applicable when one of the diverse trip features for two or more reactor trip breakers intentionally made inoperable.

ACTION 10A - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable for two or more reactor trip breakers, restore trip features on at least all but one reactor trip breaker to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours. With the diverse trip features restored to OPERABLE status on all but one reactor trip breaker, comply with ACTION 10 for the remaining inoperable trip feature on one reactor trip breaker.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection					
a. Manual Initiation	2	1	2	1 2, 3, 4	~~17~~ 28,29
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1 2, 3, 4	14
c. Containment Pressure - High	3	2	2	1 2, 3	~~15~~ 26, 27
d. Pressurizer Pressure - Low	3	2	2	1 2, 3#	~~15~~ 26, 27
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1 2, 3#	~~15~~ 26, 27

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	~~15~~ 26, 27
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2. 3*	~~15~~ 26, 27
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	~~15~~ 26, 27
Containment Pressure--High	3	2	2	1, 2, 3	~~15~~ 26, 27
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	~~17~~ 28, 29
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements. (Manual S.I. initiation will not initiate Phase A Isolation).				
b. Phase "B" Isolation					
1) Manual Initiation	2	2 (Both buttons must be pushed simultaneously to actuate)	2	1, 2, 3, 4	~~17~~ 28, 29
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	15
Containment Pressure--High	3	2	2	1, 2, 3	15
c. Containment Ventilation Isolation					
1) Containment Isolation Manual Phase A or Manual Phase B	See Items 3.a.1 and 3.b.1 above for all Manual Containment Ventilation functions and requirements.				

TABLE 3.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater### (Continued)					
b. Stm. Gen. Water Level--Low-Low	3/steam generator	2/steam generator in any steam generator	2/steam generator	1, 2, 3	~~15~~ 26, 27
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Bus Stripping	1/bus	1/bus	1/bus	1, 2, 3	23
e. Trip of all Main Feed-water Pumps Breakers	1/breaker	(1/breaker) /operating pump	(1/breaker) /operating pump	1, 2	23
7. Loss of Power					
a. 4.16 kV Busses A and B (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4	18, 18A
b. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18, 18A
Coincident with: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Loss of Power (Continued)					
c. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Degraded Voltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18, 18A
8. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure	3	2	2	1, 2, 3	19
b. T_{avg} - Low	3	2	2	1, 2, 3	19
9. Control Room Ventilation Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 6**	16
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Containment Radioactivity--High	2	1	1	1, 2, 3, 4, 6**	16
d. Containment Isolation Manual Phase A or Manual Phase B	2	1	2	1, 2, 3, 4	17
e. Control Room Air Intake Radiation Level	2	1	2	All	24

or in accordance with the Risk Informed Completion Time Program

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. ~~Both channels of any one load center may be taken out of service for up to 8 hours in order to perform surveillance testing per Specification 4.3.2.1.~~

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, comply with Specification 3.0.3.

Control Room

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the ~~control room~~ Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 25 - With number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

or in accordance with the Risk Informed Completion Time Program.

Note: Action 18A is not applicable when two channels intentionally made inoperable except both channels on any one load center may be taken out of service for up to 8 hours in order to perform surveillance testing per Specification 4.3.2.1

ACTION 18A - With the number of OPERABLE channels two less that the Total Number of Channels, restore at least one inoperable channel to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the next 30 hours. With one inoperable channel restored to OPERABLE status, comply with ACTION 18 for the remaining inoperable channel.

ACTION 26 - With one channel inoperable, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program.

Note: Action 27 is not applicable when two or more required channels intentionally made inoperable.

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ACTION 27 - With two or more required channels inoperable, restore at least all but one inoperable channel to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN with the following 6 hours. With all but one inoperable channel restored to OPERABLE status, comply with ACTION 26 for the remaining inoperable channel.

ACTION 28 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

Note: Action 29 is not applicable when a second channel is intentionally made inoperable.

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ACTION 29 - With two channels inoperable, restore at least one inoperable channel to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. With one inoperable channel restored to OPERABLE status, comply with ACTION 28 for the remaining inoperable channel.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2465 psig + 2%, -3%.* **

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** All valves tested must have "as left" lift setpoints that are within ± 1% of the lift setting value.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of boron concentration not being within limits, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

INSERT 1

b. With one accumulator inoperable due to the boron concentration not being within the limits, restore boron concentration back to the required limits within 72 hours, ~~or~~ be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the borated water volume in each accumulator is between 6520 and 6820 gallons, and

2) Verifying that the nitrogen cover pressure in each accumulator is between 600 and 675 psig, and

3) Verifying that each accumulator isolation valve is open by control room indication (power may be restored to the valve operator to perform this surveillance if redundant indicator is inoperable).

----------Note----------
Action c is not applicable when two or more accumulators intentionally made inoperable
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c. With two or more accumulators inoperable:

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1. Restore at least all but one inoperable accumulators to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

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2. With all but one inoperable accumulator restored to OPERABLE status, comply with ACTION a or b for the remaining inoperable accumulator.

Pressurizer pressure above 1000 psig.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four OPERABLE Safety Injection (SI) pumps, each capable of being powered from its associated OPERABLE diesel generator#, with discharge aligned to the RCS cold legs,*

b. Two OPERABLE RHR heat exchangers,

c. Two OPERABLE RHR pumps with discharge aligned to the RCS cold legs,

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank as defined in Specification 3.5.4, and

e. Two OPERABLE flow paths capable of taking suction from the containment sump.

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

ACTION:

a. With any one of the required ECCS components or flow paths inoperable, except for inoperable Safety Injection Pump(s) or an inoperable RHR pump, restore the inoperable component or flow path to OPERABLE status within 72 hours ~~or~~ be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 1

b. In the event the ECCS is actuated and injects water in the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date since January 1, 1990.

c. With one of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, restore the pump to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.***

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*Only three OPERABLE Safety Injection (SI) pumps (two associated with the unit and one from the opposite unit), each capable of being powered from its associated OPERABLE diesel generator#, with discharge aligned to the RCS cold leg are required if the opposite unit is in MODE 4, 5, or 6.

**The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection flow paths isolated pursuant to Specification 3.4.9.3 provided that the Safety Injection flow paths are restored to OPERABLE status prior to Tavg exceeding 380°F. Safety Injection flow paths may be isolated when Tavg is less than 380°F.

***The provisions of Specifications 3.0.4 and 4.0.4 are not applicable.

#Inoperability of the required EDG's does not constitute inoperability of the associated Safety Injection pumps.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

d. With two of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, restore one of the two inoperable pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

INSERT 1

INSERT 1

e. With one of the three required Safety Injection pumps inoperable and the opposite unit in MODE 4, 5, or 6, restore the pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. With a required Safety Injection pump OPERABLE but not capable of being powered from its associated diesel generator, restore the capability within 14 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 1

g. With an ECCS subsystem inoperable due to an RHR pump being inoperable, restore the inoperable RHR pump to OPERABLE status within 7 days or be in ~~as~~ least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 2

INSERT 1

at

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 For single Unit operation, one refueling water storage tank (RWST) shall be OPERABLE or for dual Unit operation two RWSTs shall be OPERABLE with:

a. A minimum indicated borated water volume of 320,000 gallons per RWST,

b. A boron concentration between 2400 ppm and 2600 ppm,

c. A minimum solution temperature of 39°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With less than the required number of RWST(s) OPERABLE, restore the tank(s) to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.5.4 The required RWST(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the indicated borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for consecutive 23 hours, and

2) At least once per 24 hours while the outside temperature exceeds its limit.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, or during the performance of containment air lock surveillance and/or testing requirements, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. The containment purge supply and exhaust isolation valves shall be sealed closed to the maximum extent practicable but may be open for purge system operation for pressure control, for environmental conditions control, for ALARA and respirable air quality considerations for personnel entry and for surveillance tests that require the valve to be open.

b. The purge supply and exhaust isolation valves shall not be opened wider than 33 or 30 degrees, respectively (90 degrees is fully open).

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

ACTION:

INSERT 1

a. With a containment purge supply and/or exhaust isolation valve(s) open for reasons other than given in 3.6.1.7.a above, close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Specification 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status or isolate the penetrations such that the measured leakage rate does not exceed the limits of Specification 4.6.1.7.2 within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7.a at least once per 31 days.

4.6.1.7.2 At least once per 6 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

4.6.1.7.3 At least once per 18 months, the mechanical stop on each containment purge supply and exhaust isolation valve shall be verified to be in place and that the valves will open no more than 33 or 30 degrees, respectively.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and manually transferring suction to the containment sump via the RHR System.

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

ACTION:

INSERT 1

a. With one Containment Spray System inoperable restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

b. With two Containment Spray Systems inoperable restore at least one Spray System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position and that power is available to flow path components that require power for operation;

b. By verifying that on recirculation flow, each pump develops the indicated differential pressure, when tested pursuant to Specification 4.0.5:

Containment Spray Pump $\geq$241.6 psid while aligned in recirculation mode.

EMERGENCY CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Three emergency containment cooling units shall be OPERABLE.

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

INSERT 1

a. With one of the above required emergency containment cooling units inoperable restore the inoperable cooling unit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

b. With two or more of the above required emergency containment cooling units inoperable, restore at least two cooling units to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all of the above required cooling units to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each emergency containment cooling unit shall be demonstrated OPERABLE:

a. At least once per 31 days by starting each cooler unit from the control room and verifying that each unit motor reaches the nominal operating current for the test conditions and operates for at least 15 minutes.

b. At least once per 18 months by:

1) Verifying that two emergency containment cooling units start automatically on a safety injection (SI) test signal, and

2) Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler.

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE with isolation times less than or equal to required isolation times.

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

ACTION:

*With one or more isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

INSERT 1

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic containment isolation valve secured in the isolation position, or

INSERT 1

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 The isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

*CAUTION: The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 pumps as specified in Table 3.7-3 and associated flowpaths shall be OPERABLE.

APPLICA8ILITY: MODES 1, 2 and 3

ACTION:

INSERT 1

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours, or place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours.

INSERT 1

2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify the OPERABILITY of both standby feed-water pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

INSERT 1

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours. The provisions of Specification 3.0.4 are not applicable during the 30 day period for the inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying by control panel indication and visual observation of equipment that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or

*If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

----------Note----------

The Risk Informed Completion Time Program is not applicable to Action 2 if two auxiliary feedwater trains intentionally made inoperable.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CST) system shall be OPERABLE with:

Opposite Unit in MODES 4, 5 or 6

A minimum indicated water volume of 210,000 gallons in either or both condensate storage tanks.

Opposite Unit in MODES 1, 2 or 3

A minimum indicated water volume of 420,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Opposite Unit in MODES 4, 5 or 6 INSERT 1

With the CST system inoperable, within 4 hours restore the CST system to OPERABLE status or be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Opposite Unit in MODES 1, 2 or 3 INSERT 1

1) With the CST system inoperable due to indicating less than 420,000 gallons, but greater than or equal to 210,000 gallons indicated, within 4 hours restore the inoperable CST system to OPERABLE status or place one unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 1

2) With the CST system inoperable with less than 210,000 gallons indicated, within 1 hour restore the CST system to OPERABLE status or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

INSERT 1

a. With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in ~~HOT STANDBY~~ within the next 6 hours ~~and in HOT SHUTDOWN within the following 6 hours~~.

Mode 2

MODES 2 and 3:

or more MSIVs

inoperable

With one ~~MSIV~~ inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation ~~valve is~~ maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

valves are

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

----------Note----------

Action b not applicable when two or more MSIVs intentionally made inoperable

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b. With two or more MSIVs inoperable in MODE 1, restore at least all but one inoperable MSIVs to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program or be in Mode 2 within the next 6 hours. With all but one inoperable MSIV restored to OPERABLE status, comply with ACTION a for the remaining inoperable MSIV.

3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps, and

b. Two CCW heat exchangers.

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

ACTION:

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

c. With less than two CCW heat exchangers OPERABLE, restore two heat exchangers to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE:

a. At least once per 12 hours, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.

----------Note----------
Action d is not applicable when CCW pumps intentionally made inoperable.
---

d. With three CCW pumps inoperable:

1. Restore at least one CCW pump to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one inoperable CCW pump restored to OPERABLE status, comply with ACTION b for the two remaining inoperable CCW pumps.

3/4.7.3 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The Intake Cooling Water System (ICW) shall be OPERABLE with:

a. Three ICW pumps, and
b. Two ICW headers.

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

ACTION:

INSERT 1

a. With only two ICW pumps with independent power supplies OPERABLE, restore the inoperable ICW pump to OPERABLE status within 14 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With only one ICW pump OPERABLE or with two ICW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

c. With only one ICW header OPERABLE, restore two headers to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 3 →

SURVEILLANCE REQUIREMENTS

4.7.3 The Intake Cooling Water System (ICW) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Intake Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for system operability are OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

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

ACTION:

INSERT 1

a. With one of two startup transformers or an associated circuit inoperable, demonstrate the OPERABILITY of the other startup transformer and its associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit is in MODE 1, reduce THERMAL POWER to ≤30% RATED THERMAL POWER within 24 hours, or restore the inoperable startup transformer and associated circuits to OPERABLE status within the next 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If THERMAL POWER is reduced to ≤30% RATED THERMAL POWER within 24 hours or if the inoperable startup transformer is associated with the opposite unit restore the startup transformer and its associated circuits to OPERABLE status within 30 days of the loss of OPERABILITY, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit was in MODE 2, 3, or 4 restore the startup transformer and its associated circuits to OPERABLE status within 24 hours ~~or~~ be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

INSERT 1

b. With one of the required diesel generators inoperable, demonstrate the OPERABILITY of the above required startup transformers and their associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours, unless the absence of any potential common mode failure for the remaining diesel generators is determined. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Restore the inoperable diesel generator to OPERABLE status within 14 days** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

c. With one startup transformer and one of the required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a on the remaining

** 72 hours if inoperability is associated with Action Statement 3.8.1.1.c.

ACTION (Continued)

startup transformer and associated circuits within one hour and at least once per 8 hours thereafter; and if the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable diesel generator does not exist on the remaining required diesel generators, unless the diesel generators are already operating; restore one of the inoperable sources to OPERABLE status in accordance with Action Statements a and b, as appropriate. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Notify the NRC within 4 hours of declaring both a start-up transformer and diesel generator inoperable. Restore the other A.C. power source (startup transformer or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statement a or b, as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source.

d. With one diesel generator inoperable, in addition to ACTION b. or c. above, verify that:

1. All required systems, subsystems, trains, components, and devices (except safety injection pumps) that depend on the remaining required OPERABLE diesel generators as a source of emergency power are also OPERABLE.

   If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. At least two Safety Injection pumps are OPERABLE and capable of being powered from their associated OPERABLE diesel generators.

   If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With two of the above required startup transformers or their associated circuits inoperable notify the NRC within 4 hours; restore at least one of the inoperable startup transformers to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours* and in COLD

INSERT 1

*If the opposite unit is shutdown first, this time can be extended to 42 hours.

ACTION (Continued)

SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously. With only one startup transformer and associated circuits restored, perform Surveillance Requirement 4.8.1.1.1a on the OPERABLE Startup transformer at least once per 8 hours, and restore the other startup transformer and its associated circuits to OPERABLE status or shutdown in accordance with the provisions of Action Statement 3.8.1.1a with time requirements of that Action Statement based on the time of initial loss of a startup transformer. This ACTION applies to both units simultaneously.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two startup transformers and their associated circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all required diesel generators to OPERABLE status within 14 days from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

g. Following the addition of the new fuel oil* to the Diesel Fuel Oil Storage Tanks, with one or more diesel generators with new fuel oil properties outside the required Diesel Fuel Oil Testing Program limits, restore the stored fuel oil properties to within the required limits within 30 days.

h. With one or more diesel generators with stored fuel oil total particulates outside the required Diesel Fuel Oil Testing Program limits, restore the fuel oil total particulates to within the required limits within 7 days.

----------Note----------
Action i is not applicable when three or more AC sources intentionally made inoperable.
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i. With three or more AC sources inoperable:

1. Restore at least all but two inoperable AC sources to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With all but two inoperable AC sources restored to OPERABLE status, comply with the applicable Actions for the remaining inoperable AC sources.

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* The properties of API Gravity, specific gravity or an absolute specific gravity; kinematic viscosity; clear and bright appearance; and flash point shall be confirmed to be within the Diesel Fuel Oil Testing Program limits, prior to the addition of the new fuel oil to the Diesel Fuel Oil Storage Tanks.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following D.C. electrical sources shall be OPERABLE:*#

a. 125-volt D.C. Battery Bank 3A or spare battery bank D-52 and associated full capacity charger(s)
   1) 3A1 powered by motor control center (MCC) 3C with EDG 3A OPERABLE, or
   2) 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
   3) 3A1 powered by MCC 3C with EDG 3A OPERABLE and 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

b. 125-volt D.C. Battery Bank 3B or spare battery bank D-52 and associated full capacity charger(s)
   1) 3B1 powered by MCC 3B with EDG 3B OPERABLE, or
   2) 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
   3) 3B1 powered by MCC 3B with EDG 3B OPERABLE and 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

c. 125-volt D.C. Battery Bank 4A or spare battery bank D-52 and associated full capacity charger(s)
   1) 4A1 powered by MCC 4C with EDG 4A OPERABLE, or
   2) 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or
   3) 4A1 powered by MCC 4C with EDG 4A OPERABLE and 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE,

d. 125-volt D.C. Battery Bank 4B or spare battery bank D-52 and associated full capacity charger(s)
   1) 4B1 powered by MCC 4B with EDG 4B OPERABLE, or
   2) 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or
   3) 4B1 powered by MCC 4B with EDG 4B OPERABLE and 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE.

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

ACTION:

INSERT 1

a. With one or more of the required battery chargers OPERABLE but not capable of being powered from its associated OPERABLE diesel generator(s), restore the capability within 72 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

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* All battery chargers required to satisfy the LCO shall be powered from separate MCCs.

Inoperability of the required EDG's specified in the LCO requirements below does not constitute inoperability of the associated battery chargers or battery banks.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

INSERT 1

b. With one of the required battery banks inoperable, or with none of the full-capacity chargers associated with a battery bank OPERABLE, restore all battery banks to OPERABLE status and at least one charger associated with each battery bank to OPERABLE status within two hours* or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

INSERT 4

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and its associated full capacity charger(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 129 volts on float charge and the battery charger(s) output voltage is ≥ 129 volts, and

3) If two battery chargers are connected to the battery bank, verify each battery charger is supplying a minimum of 10 amperes, or demonstrate that the battery charger supplying less than 10 amperes will accept and supply the D.C. bus load independent of its associated battery charger.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts (108.6 volts for spare battery D-52), or battery overcharge with battery terminal voltage above 143 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) The average electrolyte temperature of every sixth cell is above 60°F, and

3) There is no visible corrosion at either terminals or connectors, or verify battery connection resistance is:

Battery	Connection	Limit (Micro-Ohms)
3B, 4A	inter-cell / termination	≤ 29
	inter-cell (brace locations)	≤ 30
	transition cables	≤ 125
	or	
	total battery connections	≤ 1958
Battery	**Connection**	**Limit (Micro-Ohms)**
3A, 4B, D-52	inter-cell / termination	≤ 35
	inter-cell (brace locations)	≤ 40
	transition cables	≤ 125
	or	
	total battery connections	≤ 2463

c. At least once per 18 months by verifying that:

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

*Can be extended to 24 hours if the oppsite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

LIMITING CONDITION FOR OPERATION (Continued)

j. 120 Volt AC Vital Panel 3P09 and 3P24 energized from its associated inverter connected to D.C. Bus 4A.****

k. 120 Volt AC Vital Panel 4P09 and 4P24 energized from its associated inverter connected to D.C. Bus 4A.****

l. 125 Volt D.C. Bus 3D01 energized from an associated battery charger and from Battery Bank 3A or spare battery bank D-52,

m. 125 Volt D.C. Bus 3D23 energized from an associated battery charger and from Battery Bank 3B or spare battery bank D-52,

n. 125 Volt D.C. Bus 4D01 energized from an associated battery charger and from Battery Bank 4B or spare battery bank D-52, and

o. 125 Volt D.C. Bus 4D23 energized from an associated battery charger and from Battery Bank 4A or spare battery bank D-52

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

ACTION:

a. With one of the required trains (3.8.3.1a., b., and c) of A.C. emergency bus~~ses~~ not fully energized (except for the required LC's and MCC's associated with the opposite unit), reenergize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. INSERT 1

b. With any of the required LC's and/or MCC's associated with the opposite unit inoperable, restore the inoperable LC or MCC to OPERABLE status in accordance with Table 3.8-1 or Table 3.8-2 as applicable or place the unit in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. INSERT 1

c. With one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) Reenergize the A.C. vital panel within 2 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panel from an inverter connected to its associated D.C. bus

****A back-up inverter may be used to replace the normal inverter, provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued) INSERT 1

within 24 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

~~d~~. e With one D.C. bus not energized from its associated battery bank or associated charger, reenergize the D.C. bus from its associated battery bank within 2 hours* or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified bus~~ses~~ shall be determined energized and aligned in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

----------Note----------

Action d is not applicable when two or more A.C. vital panels intentionally either not energized from their associated inverters, or the inverters not connected to their associated D.C. buses.

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d. With two or more A.C. vital panels either not energized from their associated inverters, or with the inverters not connected to their associated D.C. buses:

.

1. Reenergize at least all but one A.C. vital panels within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously; and

.

2 Reenergize at least all but one A.C. vital panels from inverters connected to their associated D.C. buses within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

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3. With all but one A.C. vital panels energized from their associated inverters connected to their D.C. buses, comply with ACTION c for the remaining A.C vital panel either not energized from its associated inverter or with the inverter not connected to its associated D.C. bus.

* Can be extended to 24 hours if the opposite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

TABLE 3.8-1

APPLICABLE TO UNIT 3 BASED ON UNIT 4 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 4 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 3 – MODES 1, 2, 3 and 4		
	With AC Trains 3A, 3B, 4A, & 4B OPERABLE	With AC Trains 3A, 3B, & 4A OPERABLE	With AC Trains 3A, 3B, & 4B OPERABLE
LC 4A	N/A	72	N/A
MCC 4A	N/A	N/A	N/A
LC 4C and/or MCC 4C	2*	2*	N/A
LC 4H and/or MCC 4D	2**	2**	2**
LC 4B and/or MCC 4B	2*	N/A	2*
LC 4D	N/A	N/A	72

[a]

[a]

[a]

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours. INSERT 1

[a] or in accordance with the Risk Informed Completion Time Program

TABLE 3.8-2

APPLICABLE TO UNIT 4 BASED ON UNIT 3 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 3 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 4 – MODES 1, 2, 3 and 4		
	With AC Trains 4A, 4B, 3A, & 3B OPERABLE	With AC Trains 4A, 4B, & 3A OPERABLE	With AC Trains 4A, 4B, & 3B OPERABLE
LC 3A	N/A	72	N/A
LC 3C and/or MCC 3C	2*	2*	N/A
LC 3H and/or MCC 3D	2** [a]	2**	2**
LC 3B and/or MCC 3B	2*	N/A	2* [a]
LC 3D	N/A	N/A	72

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours. INSERT 1

[a] or in accordance with the Risk Informed Completion Time Program

PROCEDURES AND PROGRAMS (Continued)

3. If crack indications are found in any portion of a SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-secondary leakage.

k. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the CREVS, operating at the flow rate required by Surveillance Requirement 4.7.5.d, at a Frequency of 18 months. Additionally, the supply fans (trains A and B) will be tested on a staggered test basis (defined in Technical Specification definition 1.29 every 36 months). The results shall be trended and the CRE boundary assessed every 18 months.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by 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 Specification 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

INSERT 5

6.8.5 DELETED

ATTACHMENT 3

REVISED TECHNICAL SPECIFICATION PAGES

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1, 1A
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	13
6. Overpower ΔT	3	2	2	1, 2	13
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	13
10. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Safety Injection Input from ESF	2	1	2	1, 2	8
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2#	7
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	7
or					
Turbine Inlet Pressure	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
18. Reactor Coolant Pump Breaker Position Trip					
a. Above P-8	1/breaker	1	1/breaker	1	11
b. Above P-7 and below P-8	1/breaker	2	1/breaker	1	11
19. Reactor Trip Breakers	2	1	2	1, 2	8, 10, 10A
	2	1	2	3*, 4*, 5*	9
20. Automatic Trip and Interlock logic	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

* When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

** When the Reactor Trip System breakers are in the open position, one or both of the backup NIS instrumentation channels may be used to satisfy this requirement. For backup NIS testing requirements, see Specification 3/4.3.3.3, ACCIDENT MONITORING.

*** Reactor Coolant Pump breaker A is tripped by underfrequency sensor UF-3A1(UF-4A1) or UF-3B1(UF-4B1). Reactor Coolant Pump breakers B and C are tripped by underfrequency sensor UF-3A2(UF-4A2) or UF-3B2(UF-4B2).

\# Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

\#\# Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours.

Note: Action 1A is not applicable when second Manual Reactor Trip channel intentionally made inoperable.

ACTION 1A - With two Manual Reactor Trip channels inoperable, restore at least one channel to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours. With one channel restored to OPERABLE status, comply with ACTION 1 for the remaining inoperable channel,

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 10 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable, except for the time required for performing maintenance to restore the breaker to OPERABLE status.

Note: Action 10A is not applicable when one of the diverse trip features for two or more reactor trip breakers intentionally made inoperable.

ACTION 10A - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable for two or more reactor trip breakers, restore trip features on at least all but one reactor trip breaker to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours. With the diverse trip features restored to OPERABLE status on all but one reactor trip breaker, comply with ACTION 10 for the remaining inoperable trip feature on one reactor trip breaker.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection					
a. Manual Initiation	2	1	2	1 2, 3, 4	28, 29
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1 2, 3, 4	14
c. Containment Pressure - High	3	2	2	1 2, 3	26, 27
d. Pressurizer Pressure - Low	3	2	2	1 2, 3#	26, 27
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1 2, 3#	26, 27

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	26, 27
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2. 3*	26,27
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	26, 27
Containment Pressure--High	3	2	2	1, 2, 3	26, 27
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	28, 29
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements. (Manual S.I. initiation will not initiate Phase A Isolation).				
b. Phase "B" Isolation					
1) Manual Initiation	2	2 (Both buttons must be pushed simultaneously to actuate)	2	1, 2, 3, 4	28, 29
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15
c. Containment Ventilation Isolation					
1) Containment Isolation Manual Phase A or Manual Phase B	See Items 3.a.1 and 3.b.1 above for all Manual Containment Ventilation functions and requirements.				

TABLE 3.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater### (Continued)					
b. Stm. Gen. Water Level--Low-Low	3/steam generator	2/steam generator in any steam generator	2/steam generator	1, 2, 3	26, 27
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Bus Stripping	1/bus	1/bus	1/bus	1, 2, 3	23
e. Trip of all Main Feed-water Pumps Breakers	1/breaker	(1/breaker) /operating pump	(1/breaker) /operating pump	1, 2	23
7. Loss of Power					
a. 4.16 kV Buses A and B (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4	18, 18A
b. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18, 18A
Coincident with: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Loss of Power (Continued)					
c. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Degraded Voltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18, 18A
8. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure	3	2	2	1, 2, 3	19
b. T_{avg} - Low	3	2	2	1, 2, 3	19
9. Control Room Ventilation Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 6**	16
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Containment Radioactivity--High	2	1	1	1, 2, 3, 4, 6**	16
d. Containment Isolation Manual Phase A or Manual Phase B	2	1	2	1, 2, 3, 4	17
e. Control Room Air Intake Radiation Level	2	1	2	All	24

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program.

Note: Action 18A is not applicable when two channels intentionally made inoperable except both channels on any one load center may be taken out of service for up to 8 hours in order to perform surveillance testing per Specification 4.3.2.1.

ACTION 18A - With the number of OPERABLE channels two less that the Total Number of Channels, restore at least one inoperable channel to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the next 30 hours. With one inoperable channel restored to OPERABLE status, comply with ACTION 18 for the remaining inoperable channel.

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, comply with Specification 3.0.3.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 25 - With number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

ACTION 26 - With one channel inoperable, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program.

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

Note: Action 27 is not applicable when two or more required channels intentionally made inoperable.

ACTION 27 - With two or more required channels inoperable, restore at least all but one inoperable channel to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN with the following 6 hours. With all but one inoperable channel restored to OPERABLE status, comply with ACTION 26 for the remaining inoperable channel.

ACTION 28 With one channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the RISK Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

Note: Action 29 is not applicable when a second channel is intentionally made inoperable.

ACTION 29 With two channels inoperable, restore at least one channel to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. With one inoperable channel restored to OPERABLE status, comply with ACTION 28 for the remaining inoperable channel.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2465 psig + 2%, -3%.* **

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** All valves tested must have "as left" lift setpoints that are within ± 1% of the lift setting value.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of boron concentration not being within limits, restore the inoperable accumulator to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the boron concentration not being within the limits, restore boron concentration back to the required limits within 72 hours, or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

Note: Action c is not applicable when two or more accumulators intentionally made inoperable.

c. With two or more accumulators inoperable:

1) Restore at least all but one inoperable accumulators to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

2) With all but one inoperable accumulators restored to OPERABLE status, comply with ACTION a or b for the remaining inoperable accumulator.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the borated water volume in each accumulator is between 6520 and 6820 gallons, and

2) Verifying that the nitrogen cover pressure in each accumulator is between 600 and 675 psig, and

3) Verifying that each accumulator isolation valve is open by control room indication (power may be restored to the valve operator to perform this surveillance if redundant indicator is inoperable).

*Pressurizer pressure above 1000 psig.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four OPERABLE Safety Injection (SI) pumps, each capable of being powered from its associated OPERABLE diesel generator#, with discharge aligned to the RCS cold legs,*

b. Two OPERABLE RHR heat exchangers,

c. Two OPERABLE RHR pumps with discharge aligned to the RCS cold legs,

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank as defined in Specification 3.5.4, and

e. Two OPERABLE flow paths capable of taking suction from the containment sump.

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

ACTION:

a. With any one of the required ECCS components or flow paths inoperable, except for inoperable Safety Injection Pump(s) or an inoperable RHR pump, restore the inoperable component or flow path to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water in the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date since January 1, 1990.

c. With one of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, restore the pump to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.***

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*Only three OPERABLE Safety Injection (SI) pumps (two associated with the unit and one from the opposite unit), each capable of being powered from its associated OPERABLE diesel generator#, with discharge aligned to the RCS cold leg are required if the opposite unit is in MODE 4, 5, or 6.

**The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection flow paths isolated pursuant to Specification 3.4.9.3 provided that the Safety Injection flow paths are restored to OPERABLE status prior to Tavg exceeding 380°F. Safety Injection flow paths may be isolated when Tavg is less than 380°F.

***The provisions of Specifications 3.0.4 and 4.0.4 are not applicable.

#Inoperability of the required EDG's does not constitute inoperability of the associated Safety Injection pumps.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

d. With two of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, restore one of the two inoperable pumps to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With one of the three required Safety Injection pumps inoperable and the opposite unit in MODE 4, 5, or 6, restore the pump to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. With a required Safety Injection pump OPERABLE but not capable of being powered from its associated diesel generator, restore the capability within 14 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

g. With an ECCS subsystem inoperable due to an RHR pump being inoperable, restore the inoperable RHR pump to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Note: Action h is not applicable when two or more of the required ECCS components or flow paths intentionally made inoperable.

h. With two or more of the required ECCS components or flow paths inoperable, except for Safety Injection pumps: and RHR pumps:

1) Restore at least all but one of the inoperable components or flow paths to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT STUTDOWN within the following 6 hours.

2) With all but one inoperable ECCS component or flow path restored to OPERABLE status, comply with ACTION a for the remaining inoperable ECCS component or flow path.

Note: 1) Action i is not applicable when three or more of the required Safety Injection pumps intentionally made inoperable.

2) Action i applies to both units simultaneously.

i. With three or more of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3:

1) Restore at least all but two inoperable Safety Injection pumps to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

2) With all but two inoperable Safety Injection pumps restored to OPERABLE status, comply with ACTION d for the two remaining inoperable Safety Injection pumps.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

Note: Action j is not applicable when two or more of the required Safety Injection pumps intentionally made inoperable.

j. With two or more of the three required Safety Injection pumps inoperable and the opposite unit in MODE 4, 5, or 6:

1) Restore at least all but one inoperable Safety Injections pump to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With all but one inoperable Safety Injection pump restored to OPERABLE status, comply with ACTION e for the remaining inoperable Safety Injection pump.

Note: Action k is not applicable when two or more required Safety Injection pumps are intentionally not powered from their associated diesel generator.

k. With two or more required Safety Injection pumps OPERABLE but not capable of being powered from their associated diesel generator:

1) Restore the capability of being powered from the associated diesel generator for at least all but one Safety Injection pump within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With all but one Safety Injection pump capable of being powered from the associated diesel generator, comply with ACTION f for the remaining Safety Injection pump not capable of being powered from its associated diesel generator.

Note; Action l is not applicable when two RHR pumps intentionally made inoperable.

l. With two RHR pumps inoperable:

1) Restore at least one inoperable RHR pump to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With one inoperable RHR pump restored to OPERABLE status, comply with ACTION g for the remaining inoperable RHR pump.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 For single Unit operation, one refueling water storage tank (RWST) shall be OPERABLE or for dual Unit operation two RWSTs shall be OPERABLE with:

a. A minimum indicated borated water volume of 320,000 gallons per RWST,

b. A boron concentration between 2400 ppm and 2600 ppm,

c. A minimum solution temperature of 39°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With less than the required number of RWST(s) OPERABLE, restore the tank(s) to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The required RWST(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the indicated borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for consecutive 23 hours, and

2) At least once per 24 hours while the outside temperature exceeds its limit.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, or during the performance of containment air lock surveillance and/or testing requirements, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. The containment purge supply and exhaust isolation valves shall be sealed closed to the maximum extent practicable but may be open for purge system operation for pressure control, for environmental conditions control, for ALARA and respirable air quality considerations for personnel entry and for surveillance tests that require the valve to be open.

b. The purge supply and exhaust isolation valves shall not be opened wider than 33 or 30 degrees, respectively (90 degrees is fully open).

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

ACTION:

a. With a containment purge supply and/or exhaust isolation valve(s) open for reasons other than given in 3.6.1.7.a above, close the open valve(s) or isolate the penetration(s) within 4 hours or in accordance with the Risk Informed Completion Time Program, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Specification 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status or isolate the penetrations such that the measured leakage rate does not exceed the limits of Specification 4.6.1.7.2 within 24 hours or in accordance with the Risk Informed Completion Time Program, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7.a at least once per 31 days.

4.6.1.7.2 At least once per 6 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

4.6.1.7.3 At least once per 18 months, the mechanical stop on each containment purge supply and exhaust isolation valve shall be verified to be in place and that the valves will open no more than 33 or 30 degrees, respectively.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and manually transferring suction to the containment sump via the RHR System.

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

ACTION:

a. With one Containment Spray System inoperable restore the inoperable Spray System to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Containment Spray Systems inoperable restore at least one Spray System to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position and that power is available to flow path components that require power for operation;

b. By verifying that on recirculation flow, each pump develops the indicated differential pressure, when tested pursuant to Specification 4.0.5:

Containment Spray Pump $\geq$241.6 psid while aligned in recirculation mode.

EMERGENCY CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Three emergency containment cooling units shall be OPERABLE.

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

a. With one of the above required emergency containment cooling units inoperable restore the inoperable cooling unit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more of the above required emergency containment cooling units inoperable, restore at least two cooling units to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all of the above required cooling units to OPERABLE status within 72 hours of initial loss or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each emergency containment cooling unit shall be demonstrated OPERABLE:

a. At least once per 31 days by starting each cooler unit from the control room and verifying that each unit motor reaches the nominal operating current for the test conditions and operates for at least 15 minutes.

b. At least once per 18 months by:

1) Verifying that two emergency containment cooling units start automatically on a safety injection (SI) test signal, and

2) Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler.

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE with isolation times less than or equal to required isolation times.

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

ACTION:

*With one or more isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours or in accordance with the Risk Informed Completion Time Program, or

b. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program, by use of at least one deactivated automatic containment isolation valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program, by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 The isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

*CAUTION: The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 pumps as specified in Table 3.7-3 and associated flowpaths shall be OPERABLE.

APPLICA8ILITY: MODES 1, 2 and 3

ACTION:

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours.

Note: The Risk Informed Completion Time Program is not applicable to Action 2 if two auxiliary feedwater trains intentionally made inoperable.

2) With both required auxiliary feedwater trains inoperable, within 2 hours or in accordance with the Risk Informed Completion Time Program, either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program, verify the OPERABILITY of both standby feed-water pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours. The provisions of Specification 3.0.4 are not applicable during the 30 day period for the inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying by control panel indication and visual observation of equipment that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or

*If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CST) system shall be OPERABLE with:

Opposite Unit in MODES 4, 5 or 6

A minimum indicated water volume of 210,000 gallons in either or both condensate storage tanks.

Opposite Unit in MODES 1, 2 or 3

A minimum indicated water volume of 420,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Opposite Unit in MODES 4, 5 or 6

With the CST system inoperable, within 4 hours or in accordance with the Risk Informed Completion Time Program, restore the CST system to OPERABLE status or be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Opposite Unit in MODES 1, 2 or 3

1) With the CST system inoperable due to indicating less than 420,000 gallons, but greater than or equal to 210,000 gallons indicated, within 4 hours or in accordance with the Risk Informed Completion Time Program, restore the inoperable CST system to OPERABLE status or place one unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With the CST system inoperable with less than 210,000 gallons indicated, within 1 hour or in accordance with the Risk Informed Completion Time Program, restore the CST system to OPERABLE status or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

a. With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in Mode 2 within the next 6 hours.

Note: Action b is not applicable when two or more MSIVs intentionally made inoperable.

b. With two or more MSIVs inoperable in MODE 1, restore at least all but one inoperable MSIVs to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program or be in Mode 2 within the next 6 hours. With all but one inoperable MSIV restored to OPERABLE status, comply with ACTION a for the remaining inoperable MSIV.

MODES 2 and 3:

With one or more MSIVs inoperable, subsequent operation in MODE 2 or 3 may proceed provided the inoperable isolation valves are maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps, and

b. Two CCW heat exchangers.

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

ACTION:

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With less than two CCW heat exchangers OPERABLE, restore two heat exchangers to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Note: Action d is not applicable when CCW pumps intentionally made inoperable.

d. With three CCW pumps inoperable:

1) Restore at least one CCW pump to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2) With one inoperable CCW pump restored to OPERABLE status, comply with ACTION b for the two remaining inoperable CCW pumps.

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE:

a. At least once per 12 hours, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.

3/4.7.3 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The Intake Cooling Water System (ICW) shall be OPERABLE with:

a. Three ICW pumps, and
b. Two ICW headers.

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

ACTION:

a. With only two ICW pumps with independent power supplies OPERABLE, restore the inoperable ICW pump to OPERABLE status within 14 days or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With only one ICW pump OPERABLE or with two ICW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With only one ICW header OPERABLE, restore two headers to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Note: Action d is not applicable when ICW pumps intentionally made inoperable.

d. With three ICW pumps inoperable:

1) Restore at least one inoperable ICW pump to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2) With one inoperable ICW pump restored to OPERABLE status, comply with ACTION b for the two remaining inoperable ICW pumps.

Note: Action e is not applicable when ICW headers intentionally made inoperable.

e. With two ICW headers inoperable:

1) Restore at least one inoperable ICW header to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2) With one inoperable ICW header restored to OPERABLE status, comply with ACTION c for the remaining inoperable ICW header.

4.7.3 The Intake Cooling Water System (ICW) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Intake Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for system operability are OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

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

ACTION:

a. With one of two startup transformers or an associated circuit inoperable, demonstrate the OPERABILITY of the other startup transformer and its associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit is in MODE 1, reduce THERMAL POWER to ≤30% RATED THERMAL POWER within 24 hours, or restore the inoperable startup transformer and associated circuits to OPERABLE status within the next 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If THERMAL POWER is reduced to ≤30% RATED THERMAL POWER within 24 hours or if the inoperable startup transformer is associated with the opposite unit restore the startup transformer and its associated circuits to OPERABLE status within 30 days of the loss of OPERABILITY, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit was in MODE 2, 3, or 4 restore the startup transformer and its associated circuits to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

b. With one of the required diesel generators inoperable, demonstrate the OPERABILITY of the above required startup transformers and their associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours, unless the absence of any potential common mode failure for the remaining diesel generators is determined. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Restore the inoperable diesel generator to OPERABLE status within 14 days** or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one startup transformer and one of the required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a on the remaining

** 72 hours if inoperability is associated with Action Statement 3.8.1.1.c.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

startup transformer and associated circuits within one hour and at least once per 8 hours thereafter; and if the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable diesel generator does not exist on the remaining required diesel generators, unless the diesel generators are already operating; restore one of the inoperable sources to OPERABLE status in accordance with Action Statements a and b, as appropriate. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Notify the NRC within 4 hours of declaring both a start-up transformer and diesel generator inoperable. Restore the other A.C. power source (startup transformer or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statement a or b, as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source.

d. With one diesel generator inoperable, in addition to ACTION b. or c. above, verify that:

1. All required systems, subsystems, trains, components, and devices (except safety injection pumps) that depend on the remaining required OPERABLE diesel generators as a source of emergency power are also OPERABLE.

   If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. At least two Safety Injection pumps are OPERABLE and capable of being powered from their associated OPERABLE diesel generators.

   If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With two of the above required startup transformers or their associated circuits inoperable notify the NRC within 4 hours; restore at least one of the inoperable startup transformers to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours* and in COLD

*If the opposite unit is shutdown first, this time can be extended to 42 hours.

ACTION (Continued)

SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously. With only one startup transformer and associated circuits restored, perform Surveillance Requirement 4.8.1.1.1a on the OPERABLE Startup transformer at least once per 8 hours, and restore the other startup transformer and its associated circuits to OPERABLE status or shutdown in accordance with the provisions of Action Statement 3.8.1.1a with time requirements of that Action Statement based on the time of initial loss of a startup transformer. This ACTION applies to both units simultaneously.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two startup transformers and their associated circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all required diesel generators to OPERABLE status within 14 days from time of initial loss or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

g. Following the addition of the new fuel oil* to the Diesel Fuel Oil Storage Tanks, with one or more diesel generators with new fuel oil properties outside the required Diesel Fuel Oil Testing Program limits, restore the stored fuel oil properties to within the required limits within 30 days.

h. With one or more diesel generators with stored fuel oil total particulates outside the required Diesel Fuel Oil Testing Program limits, restore the fuel oil total particulates to within the required limits within 7 days.

Note Action i is not applicable when three or more AC sources intentionally made inoperable.

i. With three or more AC sources inoperable:

1) Restore at least all but two inoperable AC sources to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2) With all but two inoperable AC sources restored to OPERABLE status, comply with the applicable Actions for the remaining inoperable AC sources.

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* The properties of API Gravity, specific gravity or an absolute specific gravity; kinematic viscosity; clear and bright appearance; and flash point shall be confirmed to be within the Diesel Fuel Oil Testing Program limits, prior to the addition of the new fuel oil to the Diesel Fuel Oil Storage Tanks.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following D.C. electrical sources shall be OPERABLE:*#

a. 125-volt D.C. Battery Bank 3A or spare battery bank D-52 and associated full capacity charger(s)
   1) 3A1 powered by motor control center (MCC) 3C with EDG 3A OPERABLE, or
   2) 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
   3) 3A1 powered by MCC 3C with EDG 3A OPERABLE and 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

b. 125-volt D.C. Battery Bank 3B or spare battery bank D-52 and associated full capacity charger(s)
   1) 3B1 powered by MCC 3B with EDG 3B OPERABLE, or
   2) 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
   3) 3B1 powered by MCC 3B with EDG 3B OPERABLE and 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

c. 125-volt D.C. Battery Bank 4A or spare battery bank D-52 and associated full capacity charger(s)
   1) 4A1 powered by MCC 4C with EDG 4A OPERABLE, or
   2) 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or
   3) 4A1 powered by MCC 4C with EDG 4A OPERABLE and 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE,

d. 125-volt D.C. Battery Bank 4B or spare battery bank D-52 and associated full capacity charger(s)
   1) 4B1 powered by MCC 4B with EDG 4B OPERABLE, or
   2) 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or
   3) 4B1 powered by MCC 4B with EDG 4B OPERABLE and 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE.

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

ACTION:

a. With one or more of the required battery chargers OPERABLE but not capable of being powered from its associated OPERABLE diesel generator(s), restore the capability within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

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* All battery chargers required to satisfy the LCO shall be powered from separate MCCs.

Inoperability of the required EDG's specified in the LCO requirements below does not constitute inoperability of the associated battery chargers or battery banks.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. With one of the required battery banks inoperable, or with none of the full-capacity chargers associated with a battery bank OPERABLE, restore all battery banks to OPERABLE status and at least one charger associated with each battery bank to OPERABLE status within two hours* or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

Note: 1) Action c is not applicable when two or more battery banks or associated full capacity chargers intentionally made inoperable.

2) Action c applies to both units simultaneously.

c. With two or more battery banks inoperable or with the full capacity chargers associated with two or more battery banks inoperable:

1) Restore at least all but one inoperable battery bank to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours, and

2) Restore one full-capacity battery charger associated with at least all but one battery bank to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours

3) With all but one inoperable battery bank restored to OPERABLE status, comply with ACTION b for the inoperable battery bank.

4) With all but one battery bank having at least one of its associated full-capacity battery chargers restored to OPERABLE status, comply with ACTION b for the inoperability of all battery chargers associated with the battery bank.

*Can be extended to 24 hours if the opposite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and its associated full capacity charger(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 129 volts on float charge and the battery charger(s) output voltage is ≥ 129 volts, and

3) If two battery chargers are connected to the battery bank, verify each battery charger is supplying a minimum of 10 amperes, or demonstrate that the battery charger supplying less than 10 amperes will accept and supply the D.C. bus load independent of its associated battery charger.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts (108.6 volts for spare battery D-52), or battery overcharge with battery terminal voltage above 143 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) The average electrolyte temperature of every sixth cell is above 60°F, and

3) There is no visible corrosion at either terminals or connectors, or verify battery connection resistance is:

Battery	Connection	Limit (Micro-Ohms)
3B, 4A	inter-cell / termination	≤ 29
	inter-cell (brace locations)	≤ 30
	transition cables	≤ 125
	or	
	total battery connections	≤ 1958
Battery	**Connection**	**Limit (Micro-Ohms)**
3A, 4B, D-52	inter-cell / termination	≤ 35
	inter-cell (brace locations)	≤ 40
	transition cables	≤ 125
	or	
	total battery connections	≤ 2463

c. At least once per 18 months by verifying that:

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

LIMITING CONDITION FOR OPERATION (Continued)

j. 120 Volt AC Vital Panel 3P09 and 3P24 energized from its associated inverter connected to D.C. Bus 4A.****

k. 120 Volt AC Vital Panel 4P09 and 4P24 energized from its associated inverter connected to D.C. Bus 4A.****

l. 125 Volt D.C. Bus 3D01 energized from an associated battery charger and from Battery Bank 3A or spare battery bank D-52,

m. 125 Volt D.C. Bus 3D23 energized from an associated battery charger and from Battery Bank 3B or spare battery bank D-52,

n. 125 Volt D.C. Bus 4D01 energized from an associated battery charger and from Battery Bank 4B or spare battery bank D-52, and

o. 125 Volt D.C. Bus 4D23 energized from an associated battery charger and from Battery Bank 4A or spare battery bank D-52

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

ACTION:

a. With one of the required trains (3.8.3.1a., b., and c) of A.C. emergency buses not fully energized (except for the required LC's and MCC's associated with the opposite unit), reenergize the train within 8 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any of the required LC's and/or MCC's associated with the opposite unit inoperable, restore the inoperable LC or MCC to OPERABLE status in accordance with Table 3.8-1 or Table 3.8-2 as applicable or place the unit in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) Reenergize the A.C. vital panel within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panel from an inverter connected to its associated D.C. bus

****A back-up inverter may be used to replace the normal inverter, provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

Note: Action d is not applicable when two or more A.C. vital panels intentionally either not energized from their associated inverters, or the inverters not connected to their associated D.C buses.

d. With two or more A.C vital panels either not energized from their associated inverters, or with the inverters not connected to their associated D.C buses:

1) Reenergize at least all but one A.C. vital panels within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously; and

2) Reenergize at least all but one A.C. vital panels from inverters connected to their associated D.C.buses within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

3) With all but one AC vital panels energized from their associated inverters connected to their D.C. buses, comply with ACTION c for the remaining A.C vital panel either not energized from its associated inverter or with the inverter not connected to its associated D.C. bus.

e. With one D.C. bus not energized from its associated battery bank or associated charger, reenergize the D.C. bus from its associated battery bank within 2 hours* or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified buses shall be determined energized and aligned in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the buses.

* Can be extended to 24 hours if the opposite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

TABLE 3.8-1

APPLICABLE TO UNIT 3 BASED ON UNIT 4 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 4 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 3 – MODES 1, 2, 3 and 4		
	With AC Trains 3A, 3B, 4A, & 4B OPERABLE	With AC Trains 3A, 3B, & 4A OPERABLE	With AC Trains 3A, 3B, & 4B OPERABLE
LC 4A	N/A	72[a]	N/A
MCC 4A	N/A	N/A	N/A
LC 4C and/or MCC 4C	2*[a]	2*[a]	N/A
LC 4H and/or MCC 4D	2**[a]	2**[a]	2**[a]
LC 4B and/or MCC 4B	2*[a]	N/A	2*[a]
LC 4D	N/A	N/A	72[a]

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours or in accordance with the Risk Informed Completion Time Program.

[a] or in accordance with the Risk Informed Completion Time Program.

TABLE 3.8-2

APPLICABLE TO UNIT 4 BASED ON UNIT 3 LOAD
CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 3 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 4 – MODES 1, 2, 3 and 4		
	With AC Trains 4A, 4B, 3A, & 3B OPERABLE	With AC Trains 4A, 4B, & 3A OPERABLE	With AC Trains 4A, 4B, & 3B OPERABLE
LC 3A	N/A	72[a]	N/A
LC 3C and/or MCC 3C	2*[a]	2*[a]	N/A
LC 3H and/or MCC 3D	2**[a]	2**[a]	2**[a]
LC 3B and/or MCC 3B	2*[a]	N/A	2*[a]
LC 3D	N/A	N/A	72[a]

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours or in accordance with the Risk Informed Completion Time Program.

[a] or in accordance with the Risk Informed Completion Time Program.

3. If crack indications are found in any portion of a SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-secondary leakage.

k. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the CREVS, operating at the flow rate required by Surveillance Requirement 4.7.5.d, at a Frequency of 18 months. Additionally, the supply fans (trains A and B) will be tested on a staggered test basis (defined in Technical Specification definition 1.29 every 36 months). The results shall be trended and the CRE boundary assessed every 18 months.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by 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 Specification 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

PROCEDURES AND PROGRAMS (Continued)

l. 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, "Risk-Informed Technical Specification Initiative 4b: Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, November 2006. The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 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.

6.8.5 DELETED

ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES (MARKUPS)

ATTACHMENT 2
Technical Specification Bases
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3/4.3.1 & 3/4.3.2 (Continued)

The Engineered Safety Features Actuation System interlocks perform the following functions:

- HIGH STEAM FLOW SAFETY INJECTION BLOCK - This permissive is used to block the safety injection (SI) signal generated by High Steam Line Flow coincident with Low Steam Line Pressure or Low Tavg. The permissive is generated when two out of three Low Tavg channels drop below their setpoints and the manual SI Block/Unblock switch is momentarily placed in the block position. This switch is a spring return to the normal position type. The permissive will automatically be defeated if two out of three Low Tavg channels rise above their setpoints. The permissive may be manually defeated when two out of three Low Tavg channels are below their setpoints and the manual SI Block/Unblock switch is momentarily placed in the unblock position.

- LOW PRESSURIZER PRESSURE SAFETY INJECTION BLOCK - This permissive is used to block the safety injection signals generated by Low Pressurizer Pressure and High Differential Pressure between the Steam Line Header and any Steam Line. The permissive is generated when two out of three pressurizer pressure permissive channels drop below their setpoints and the manual SI Block/Unblock switch is momentarily placed in the block position. This is the same switch that is used to manually block the High Steam Flow Safety Injection signals mentioned above. This permissive will automatically be defeated if two out of three pressurizer pressure permissive channels rise above their setpoints. The permissive may be manually defeated when two out of three pressurizer pressure permissive channels are below their setpoints and the manual SI Block/Unblock switch momentarily placed in the Unblock position.

Action Statements

Action 1 - With one Manual Reactor Trip channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program.

Action 1A - With two Manual Reactor Trip channels inoperable the Action is to restore the inoperable channels to OPERABLE status within one hour. The one hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of the inoperable channels. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

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

Action 10 - With one of the reactor trip breakers (RTB) diverse trip features (undervoltage or shunt trip) inoperable, it must be restored to an OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program.

Action 10A - With one trip mechanism inoperable for two or more RTBs, the Action is to restore all but one inoperable trip mechanism to OPERABLE status within one hour. The one hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of the trip mechanisms. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

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

Action 18 - With the number of OPERABLE channels for loss of power instrumentation (functional units 7.a, 7.b, 7.c) one less than the Total Number of Channels, the Action requires placing the inoperable channel in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program.

Action 18A - With the number of OPERABLE channels for loss of power instrumentation (functional units 7.a, 7.b, 7.c) two less than the Total Number of Channels, the Action requires restoring the channels to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program.

The Action is modified by a Note stating it is not applicable when the two channels are intentionally made inoperable except both channels on any one load center may be removed from service for up to eight hours to perform surveillance testing per TS 4.3.2.1. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if one channel is inoperable for any reason and the second channel is found to be inoperable, or if both channels are found to be inoperable at the same time.

Action 25 - With one less than the total number of channels OPERABLE, the Action requires placing the inoperable channel in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program.

Action 26 - With one channel inoperable, the Action requires placing the inoperable channel in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program.

Action 27 - With two or more channels inoperable or one channel inoperable in more than one loop, steam line, or steam generator, the Action is to restore sufficient channels to OPERABLE status within one hour. The one hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of sufficient channels. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Action is modified by a Note stating it is not applicable when two or more required channels or one channel in more than one loop, steam line, or steam generator are intentionally made inoperable. The Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if one channel is inoperable for any reason and additional channels are found to be inoperable, or if two or more channels are found to be inoperable at the same time.

Action 28 - With one channel of Manual Initiation inoperable, the Acton requires restoring the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program.

Action 29 - With two channels of Manual Initiation inoperable, the Action is to restore the inoperable channels to OPERABLE status within one hour. The one hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of the inoperable channels. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

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

ATTACHMENT 2
Technical Specification Bases
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3/4.4.1 (Continued)

The Technical Specifications for Cold Shutdown allow an inoperable RHR pump to be the operating RHR pump for up to 2 hours for surveillance testing to establish operability. This is required because of the piping arrangement when the RHR system is being used for Decay Heat Removal.

3/4.4.2 Safety Valves

The Pressurizer Code Safety Valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 313,826 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that **NO** safety valves are OPERABLE, an RCS vent opening of at least 2.20 square inches will provide overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Mitigating System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all Pressurizer Code Safety Valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming **NO** Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., **NO** credit is taken for a direct Reactor trip on the loss-of-load) and also assuming **NO** operation of the power-operated relief valves or steam dump valves. *With one pressurizer safety valve inoperable, restoration must take place within 15 minutes or in accordance with the Risk Informed Completion Time Program.*

The pressurizer safety valves are set to open at an RCS pressure of 2465 psig +2% and -3% to avoid exceeding the maximum design pressure safety limit and to maintain accident assumptions. The pressurizer safety valve lift setting is needed to assure acceptable results for the Loss of Load/ Turbine Trip analysis. The upper and lower pressure tolerance limits are based on the tolerance requirements assumed in the safety analyses.

ATTACHMENT 2
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3/4.5.1 (Continued)

For an Accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the surveillance requirements for contained volume, boron concentration, and nitrogen cover pressure must be met. Operability of the accumulators does **NOT** depend on the Operability of the water level and pressure channel instruments, therefore, Accumulator volume and nitrogen cover pressure surveillance may be verified by any valid means, **NOT** just by instrumentation.

If the boron concentration of one Accumulator is **NOT** within limits, it must be returned to within the limits within 72 hours *or in accordance with the Risk Informed Completion Time Program.* ~~.~~ In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the Accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break Loss of Coolant Accident (LOCA) is sufficient to keep that portion of the core subcritical. One Accumulator below the minimum boron concentration limit, however, will have **NO** effect on available ECCS water and an insignificant effect on core subcriticality during reflood. In addition, current Turkey Point analysis demonstrates that the Accumulators discharge only a small amount following a large main steam line break. Their impact is minor since the use of the Accumulator volume compensates for Reactor Coolant System shrinkage and the change in boron concentration is insignificant. Thus, 72 hours is allowed to return the boron concentration to within limits.

If one Accumulator is inoperable for a reason other than boron concentration, the Accumulator must be returned to OPERABLE status within 1 hour *or in accordance with the Risk Informed Completion Time Program.*. In this condition, the required contents of three Accumulators 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 1 hour completion time to open the valve, remove power to the valve, or restore the 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 potential for exposure of the plant to a LOCA under these conditions.

With two or more accumulators inoperable, the required action is to restore sufficient inoperable accumulators to OPERABLE status within one hour to regain this safety function. The one-hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of sufficient accumulators. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

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

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3/4.5.2 & 3/4.5.3 (Continued)

When PC-600/-601 are calibrated, a test signal is supplied to each circuit to check operation of the relays and annunciators operated by subject controllers. This test signal will prevent MOVs 862A, 862B, 863A, 863B from opening. Therefore, it is appropriate to tag out the MOV breakers, and enter Technical Specification Action Statement 3.5.2.a. and 3.6.2.1 when calibrating PC-600/-601.

With the RCS temperature below 350ůF, operation with less than full redundant equipment is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

TS 3.5.2, Action g. provides an allowed outage/action completion time (AOT) of up to 7 days *or in accordance with the Risk Informed Completion Time Program* to restore an inoperable RHR Pump to OPERABLE status, provided the affected ECCS Subsystem is inoperable only because its associated RHR Pump is inoperable. This 7 day AOT is based on the results of a deterministic and probabilistic safety assessment, and is referred to as a Risk-Informed AOT Extension. Planned entry into this AOT requires that a Risk Assessment be performed in accordance with the Configuration Risk Management Program (CRMP), which is described in the administrative procedure that implements the Maintenance Rule pursuant to 10CFR50.65.

TS 3.5.2, Action h

With two or more of the required ECCS components or flow paths inoperable, except for Safety Injection Pumps and RHR pumps, Action h requires restoring the inoperable components or flow paths to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program. The Action is modified by a Note stating it is not applicable when two or more of the required ECCS components or flow paths are intentionally made inoperable. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if one required ECCS component or flow path is inoperable for any reason and a second required ECCS component or flow path is found inoperable, or if two or more required ECCS components or flow paths are found inoperable at the same time.

TS 3.5.2. Action i

With three or more of the required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, Action i requires restoring Safety Injection pumps to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program. The Action is modified by a Note stating it is not applicable when three or more of the required Safety Injection pumps are intentionally made inoperable. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if two Safety Injection pumps are inoperable for any reason and additional Safety Injection pumps are found inoperable, or if three or more Safety Injection pumps are found inoperable at the same time.

TS 3.5.2. Action j

With two or more of the required Safety Injection pumps inoperable and the opposite unit in MODE 4, 5, or 6, Action j requires restoring Safety Injection pumps to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program. The Action is modified by a Note stating it is not applicable when two or more of the required Safety Injection pumps are intentionally made inoperable. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if one Safety Injection pump is inoperable for any reason and additional Safety Injection pumps are found inoperable, or if two or more Safety Injection pumps are found inoperable at the same time.

TS 3.5.2. Action k

With two or more required Safety Injection pumps OPERABLE but not capable of being powered from their associated diesel generator, Action k requires restoring capability within two hours or in accordance with the Risk Informed Completion Time Program. The Action is modified by a Note stating it is not applicable when two or more required Safety Injection pumps intentionally not powered from their associated diesel generator. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if one Safety Injection pump is not capable of being powered from its associated diesel generator for any reason and additional Safety Injection pumps are found incapable of being powered from their associated diesel generator, or if two or more Safety Injection pumps are found incapable at the same time.

TS 3.5.2. Action l

With two RHR pumps inoperable, Action l requires restoring the RHR pumps to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program. The Action is modified by a Note stating it is not applicable when two RHR pumps are intentionally made inoperable. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if one RHR pump is inoperable for any reason and the other RHR pump is found inoperable, or if two RHR pumps are found inoperable at the same time.

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3/4.5.4 Refueling Water Storage Tank

Pump performance requirements are obtained from accident analysis assumptions. Varying flowrates are provided to accommodate testing during modes and alignments.

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) Sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) The reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods assumed out of the core to maximize boron requirements. *With less than the required number of RWSTs OPERABLE, the inoperable RWST(s) must be returned to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program.*

The assumptions made in the LOCA analyses credit control rods for the SBLOCA and cold leg large break LOCA and do **NOT** credit control rods for the hot leg large break LOCA. For the Cold Leg Large Break LOCA, control rods are assumed inserted only at the time of hot leg switchover to provide the additional negative reactivity required to address concerns of potential core recriticality at the time. (Reference: PTN-ENG-SEFJ-02-016 approved 11/14/03, PNSC #03-167.)

The indicated water volume limit includes an allowance for water **NOT** usable because of tank discharge line location or other physical characteristics.

The temperature limits on the RWST solution ensure that:
1) The solubility of the borated water will be maintained, and
2) The temperature of the RWST solution is consistent with the LOCA analysis. Portable instrumentation may be used to monitor the RWST temperature.

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3/4.6.2.1 (Continued)

The allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment and do **NOT** reflect the additional redundancy in cooling capability provided by the Emergency Containment Cooling System. *Alternatively, a completion time can be determined in accordance with the Risk Informed Completion Time Program.* Pump performance requirements are obtained from the accidents analysis assumptions.

Motor Operated Valves (MOVs) 862A, 862B, 863A, 863B are required to take suction from the Containment Sump via the RHR system. PC-600 supplies controlling signals to valves MOVs 862B and 863B, to prevent opening these valves if RHR pump B discharge pressure is above 210 psig. PC-601 provides similar functions to valves MOVs 862A and 863A. Although all four valves are normally locked in position, with power removed, the capability to power up and stroke the valves must be maintained in order to satisfy the requirements for OPERABLE flow paths (capable of taking suction from the containment sump).

When PC-600/-601 are calibrated, a test signal is supplied to each circuit to check operation of the relays and annunciators operated by subject controllers. This test signal will prevent MOVs 862A, 862B, 863A, 863B from opening. Therefore, it is appropriate to tag out the MOV breakers, and enter Technical Specification Action Statement 3.5.2.a. and 3.6.2.1 when calibrating PC-600/-601.

3/4.6.2.2 Emergency Containment Cooling System

The OPERABILITY of the Emergency Containment Cooling (ECC) System ensures that the heat removal capacity is maintained with*in* acceptable ranges following postulated design basis accidents. To support both containment integrity safety analyses and component cooling water thermal analysis, a maximum of two ECC *units* can receive an automatic start signal following generation of a safety injection (SI) signal (one ECC *unit* receives an A train SI signal and another ECC *unit* receives a B train SI signal). To support post-LOCA long-term containment pressure/temperature analyses, a maximum of two ECC *units* are required to operate. The third (swing) ECC *unit* is required to be OPERABLE to support manual starting following a postulated LOCA event for containment pressure/temperature suppression.

3/4.6.2.2 (Continued)

The allowable out-of-service time requirements for the Containment Cooling System have been maintained consistent with that assigned other inoperable ESF equipment and do **NOT** reflect the additional redundancy in cooling capability provided by the Containment Spray System. *Alternatively, a completion time can be determined in accordance with the Risk Informed Completion Time Program.*

The surveillance requirement for ECC flow is verified by correlating the test configuration value with the design basis assumptions for system configuration and flow. An 18-month surveillance interval is acceptable based on the use of water from the CCW system, which results in a low risk of heat exchanger tube fouling.

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3/4.7.1.2 Auxiliary Feedwater System

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total Loss-Of-Offsite Power. Steam can be supplied to the pump turbines from either or both units through redundant steam headers. Two D.C. motor operated valves and one A.C. motor operated valve on each unit isolate the three main steam lines from these headers. Both the D.C. and A.C. motor operated valves are powered from safety-related sources. Auxiliary feedwater can be supplied through redundant lines to the safety-related portions of the main feedwater lines to each of the steam generators. Air operated fail closed flow control valves are provided to modulate the flow to each steam generator. Each Steam Driven Auxiliary Feedwater Pump has sufficient capacity for single and two unit operation to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

ACTION statement 2 describes the actions to be taken when both Auxiliary Feedwater Trains are inoperable. *The Action is modified by a note that states the Risk Informed Completion Time (RICT) Program is not applicable if two Auxiliary Feedwater Trains are intentionally made inoperable. A RICT is not intended for voluntary removal of redundant systems or components from service. A RICT is only applicable if one Auxiliary Feedwater Train is inoperable for any reason and the other Auxiliary Feedwater Train is found inoperable or both Auxiliary Feedwater Trains are found inoperable at the same time.* The requirement to verify the availability of both Standby Feedwater Pumps is to be accomplished by verifying that both pumps have successfully passed their monthly surveillance tests within the last surveillance interval. The requirement to complete this action before beginning a unit shutdown is to ensure that an alternate feedwater train is available before putting the affected unit through a transient. If **NO** alternate feedwater trains are available, the affected unit is to stay at the same condition until an auxiliary feedwater train is returned to service, and then invoke ACTION statement 1 for the other train. If both Standby Feedwater Pumps are made available before one Auxiliary Feedwater Train is returned to an OPERABLE status, then the affected units shall be placed in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.

3/4.7.1.5 Main Steam Line Isolation Valves

The OPERABILITY of the Main Steam Line Isolation Valves (MSIV) ensures that **NO** more than one Steam Generator will blow down in the event of a steam line rupture. This restriction is required to: (1) Minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) Limit the pressure rise within Containment in the event a Main Steam Line or Feedwater Line rupture occurs within Containment. The OPERABILITY of the Main Steam Isolation Valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the Safety Analyses. *With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program.* The 24 hour ACTION time provides a reasonable amount of time to troubleshoot and repair the system.

Action a

With one MSIV inoperable and open in MODE 1, the Action requires restoring the inoperable MSIV to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program.

Action b

With two or more MSIVs inoperable in MODE 1, the required action is to restore a sufficient number of inoperable MSIVs to OPERABLE status within one hour to regain a method of main steam line isolation. The one-hour completion time is acceptable because it minimizes risk while allowing time for restoration of sufficient MSIVs. Alternatively, a completion time can be determined in accordance with the Risk Informed Completion time Program.

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

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3/4.7.2 Component Cooling Water System

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single active failure, is consistent with the assumptions used in the safety analyses. One pump and two heat exchangers provide the heat removal capability for accidents that have been analyzed.

Action c

With less than two CCW heat exchangers OPERABLE, the required Action is to restore two heat exchangers to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program.

Action d

With less than one CCW pump operable, the required action is to restore the inoperable CCW pumps to OPERABLE status within one hour to regain a heat sink for safety related components. The one-hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one pump. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

Action d is modified by a Note stating it is not applicable when the CCW pumps are intentionally made inoperable. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if two CCW pumps are inoperable for any reason and a third CCW pump is found to be inoperable, or if three CCW pumps are found to be inoperable at the same time

3/4.7.3 Intake Cooling Water System

The OPERABILITY of the Intake Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The design and operation of this system, assuming a single active failure, ensures cooling capacity consistent with the assumptions used in the safety analyses.

Action d

With three ICW pumps inoperable, the required action is to restore the inoperable ICW pumps to OPERABLE status within one hour to regain a heat sink for safety related components. The one-hour completion time is acceptable because it minimizes risk while allowing time for restoration of at least one train. Alternatively, a completion time can be determined in accordance with the Risk Informed Completion Time Program.

Action d is modified by a Note stating it is not applicable when the ICW pumps are intentionally made inoperable. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if two ICW pumps are inoperable for any reason and a third ICW pump is found to be inoperable, or if three ICW pumps are found to be inoperable at the same time.

Action e

With two ICW headers inoperable, the required action is to restore the inoperable ICW headers to OPERABLE status within one hour to regain a heat sink for safety related components. The one-hour completion time is acceptable because it minimizes risk while allowing time for restoration of at least one train. Alternatively, a completion time can be determined in accordance with the Risk Informed Completion Time Program.

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

3/4.7.4 Ultimate Heat Sink

The limit on Ultimate Heat Sink (UHS) temperature in conjunction with the SURVEILLANCE REQUIREMENTS of Technical Specification 3/4.7.2 will ensure that sufficient cooling capacity is available either: (1) To provide normal cooldown of the facility, or (2) To mitigate the effects of accident conditions within acceptable limits.

FPL has the option of monitoring the UHS temperature by monitoring the temperature in the ICW System piping going to the inlet of the CCW Heat Exchangers. Monitoring the UHS temperature after the ICW but prior to CCW Heat Exchangers is considered to be equivalent to temperature monitoring before the ICW Pumps. The supply water leaving the ICW Pumps will be mixed and therefore, it will be representative of the bulk UHS temperature to the CCW Heat Exchanger inlet. The effects of the pump heating on the supply water are negligible due to low ICW head and high water volume. Accordingly, monitoring the UHS temperature after the ICW Pumps but prior to the CCW Heat Exchangers provides an equivalent location for monitoring the UHS temperature.

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3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

All diesel generator inoperabilities must be investigated for Common Cause Failures regardless of how long the diesel generator inoperability persists. When one diesel generator is inoperable, TS 3.8.1.1 ACTION statements b and c provide an allowance to avoid unnecessary testing of other required diesel generators. If it can be determined that the cause of the inoperable diesel generator does **NOT** exist on the remaining required diesel generators, then SR 4.8.1.1.2a.4 does **NOT** have to be performed. Twenty-four (24) hours (or eight (8) hours if both a Startup Transformer and diesel generator are inoperable) is reasonable to confirm that the remaining required diesel generators are **NOT** affected by the same problem as the inoperable diesel generator. When an EDG itself is inoperable (**NOT** including a support system or independently testable component), the other EDGs should be tested once unless the absence of any potential common-mode failure can be demonstrated. If it cannot otherwise be determined that the cause of the initial inoperable diesel generator does **NOT** exist on the remaining required diesel generators, then satisfactory performance of SR 4.8.1.1.2a.4 suffices to provide assurance of continued OPERABILITY of the remaining required diesel generators. If the cause of the initial inoperability exists on one or more of the remaining required diesel generators, those diesel generators affected would also be declared inoperable upon discovery, and TS 3.8.1.1 ACTION statement f or ~~TS 3.0.3~~*ACTION i*, as appropriate, would apply.

When in Modes 1, 2, 3 or 4, a unit depends on one EDG and its associated train of busses from the opposite unit in order to satisfy the single active failure criterion for safety injection (SI) pumps and other shared equipment required during a loss-of-coolant accident with a loss-of-offsite power. Therefore, one EDG from the opposite unit is required to be OPERABLE along with the two EDGs associated with the applicable unit.

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3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

For single unit operation (one unit in MODES 1-4 and one unit in Modes 5-6 or defueled) TS 3.8.1.1 ACTION d. refers to one of the three required Emergency Diesel Generators. For dual unit operation (both units in MODES 1-4), TS 3.8.1.1 ACTION d. refers to one of the four required Emergency Diesel Generators. This conclusion is based on the portion of ACTION d. that states "in addition to ACTION b. or c" Since ACTIONs b. and c. both refer to one of the required diesel generators, this implies that ACTION d. also refers to one of the required diesel generators. ACTION d. says "in addition to ACTION b. or c. above, ..." therefore, ACTION d. is merely providing additional requirements applicable to the conditions that required satisfaction of ACTIONs b. or c.

TS 3.8.1.1. Action e

With both Startup Transformers inoperable, the units are required to be shutdown consecutively, after 24 hours *in accordance with Action e. Alternatively, a completion time can be determined in accordance with the Risk Informed Completion Time Program.* A consecutive shutdown is used because a unit without its associated transformer must perform a natural circulation cooldown. By placing one unit in COLD SHUTDOWN before starting shutdown of the second unit, a dual unit natural circulation cooldown is avoided.

The term verify means to administratively check by examining logs or other information to determine if required components are out-of-service for maintenance or other reasons. It does **NOT** mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

TS 3.8.1.1. Action i

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

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

In accordance with Technical Specification Amendments 215/209 during MODES 1, 2, and 3, if an EDG is to be removed from service for maintenance for a period scheduled to exceed 72 hours, the following restrictions apply:

If an EDG is unavailable, the Startup Transformer will be removed from service only for corrective maintenance, i.e., maintenance required to ensure or restore operability.

If the Startup Transformer is unavailable, an EDG will be removed from service only for corrective maintenance, i.e., maintenance required to ensure or restore operability.

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3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

The frequency for performing surveillance on stored fuel oil is based on stored fuel oil degradation trends which indicate that particulate concentration is unlikely to change significantly between surveillances.

The OPERABILITY of the minimum specified A.C. and D.C. Power Sources and associated distribution systems during shutdown and refueling ensures that (1) The facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

During a unit shutdown, the one required circuit between the offsite transmission network and the onsite Class 1E Distribution System can consist of at least the associated unit startup transformer feeding one 4160 volt Bus A or B, or the opposite unit's startup transformer feeding the associated unit's 4160 volt Bus A, or the associated unit's 4160 volt Bus A or B backfed through its auxiliary transformers with the main generator isolated.

As inoperability of numerous electrical components often affects the operation of the opposite unit, the applicability for the shutdown LIMITING CONDITION FOR OPERATION (LCO) for A.C. Sources, D.C. Sources and Onsite Power Distribution all contain statements to ensure the LCOs of the opposite unit are considered.

The allowable out-of-service time for the D.C. buses is 24 hours with one unit shutdown in order to allow for required battery maintenance without requiring both units to be shutdown. *Alternatively, a completion time can be determined in accordance with the Risk Informed Completion Time Program.* Provisions to substitute the spare battery for any one of the four station batteries have been included to allow for battery maintenance without requiring both units to be shutdown. The requirement to have only one OPERABLE battery charger associated with a required battery bank permits maintenance to be conducted on the redundant battery charger.

3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

The minimum number of battery chargers required to be OPERABLE is based on the following criteria:

1) A minimum of one battery charger per bus with each powered from a separate 480 volt MCC is required to satisfy the single failure criteria when assuming the failure of a MCC. This restriction prohibits the use of two chargers powered from the same bus for meeting the minimum requirements.

2) To satisfy the single failure criteria, when assuming a Loss-Of-Offsite Power with the loss of an EDG, an additional restriction is stipulated which requires each battery charger to have its associated diesel generators OPERABLE. This requires both EDGs associated with a swing bus battery charger to be OPERABLE.

Provisions for requiring the OPERABILITY of the EDG associated with the battery charger is explicitly specified in the LCO. This is because conditions exist where the affected unit would **NOT** enter the applicable ACTION statement in the LCO without this provision. For example, with Unit 3 in MODE 1 and Unit 4 in MODE 5, the operability of both EDG 4A and 4B is **NOT** required. One could postulate conditions where battery chargers 4A1, 3A2, 3B2, or 4B1 could be used to satisfy the LCO without having an associated OPERABLE EDG, unless specific provisions were made to preclude these conditions.

An out-of-service limit of 72 hours is applied when the required EDG is **NOT** OPERABLE. With less than the required battery chargers OPERABLE, an allowable out-of-service time of 2 hours is applied, which can be extended to 24 hours if the opposite unit is in MODES 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generators. *Alternatively, completion times can be determined in accordance with the Risk Informed Completion Time Program.*

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensure the effectiveness of the charging system, the ability to handle high discharge rates, and verifies the battery capability to supply its required load.

TS 3.8.2.1, Action c

With two or more battery banks inoperable or with the full capacity chargers associated with two or more battery banks inoperable, the required Action is to restore at least one of the required inoperable battery bank and associated charger to OPERABLE status within one hour to regain control power for the AC emergency power system. The one-hour completion time is acceptable because it minimizes risk while allowing time for restoration of at least one DC electrical power train. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

Action c is modified by a Note stating it is not applicable when two or more battery banks or associated full capacity chargers are intentionally made inoperable. This required action is not intended for voluntary removal of redundant systems or components from service. Action c is only applicable if one battery bank or full capacity charger is inoperable for any reason and one or more batteries or chargers are found to be inoperable, or if two or more batteries or chargers are found to be inoperable at the same time.

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensure the effectiveness of the charging system, the ability to handle high discharge rates, and verifies the battery capability to supply its required load.

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3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

The ACTION requirements specified for the inoperability of certain Motor Control Centers (MCCs), Load Centers (LCs) and the 4160-Volt Busses provide restrictions upon continued facility operation commensurate with the level of degradation on each unit and the amount of time one could reasonably diagnose and correct a minor problem. The level of degradation is based upon the types of equipment powered and the out-of-service limit imposed on that equipment by the associated ACTION statement. If this degradation affects the associated unit only, then **NO** restriction is placed on the opposite unit and an out-of-service limit of 8 hours (except for MCCs 3A, 3K, 4J and 4K) is applied to the associated unit. Since MCCs 3A, 3K, 4J and 4K are used to power EDG auxiliaries, an out-of-service limit of 72 hours is applied as required by 3.8.1.1. If the degradation impacts both units (i.e., required shared systems or cross-unit loads), then an out-of-service limit of 8 hours is applied to the associated unit and an out-of-service limit based on the most restrictive ACTION requirement for the applicable shared or cross- unit load is applied to the opposite unit. *Alternatively, completion times can be determined in accordance with the Risk Informed Completion Time Program.*

For example, if being used to satisfy 3.8.2.1, the Battery Chargers 3A2, 3B2, 4A2, and 4B2 are cross-unit loads and have out-of-service limits of 2 hours. This is the most restrictive limit of the applicable equipment powered from MCC 3D and 4D. Therefore, an out of service limit of 2 hours is applied if the battery charger is required to be OPERABLE. *Alternatively, completion times can be determined in accordance with the Risk Informed Completion Time Program.*

TS 3.8.3.1, Action c

The ACTION requirements specified when an A.C. vital panel is **NOT** energized from an inverter connected to its associated D.C. bus provides for two phases of restoration. Expedient restoration of an A.C. panel is required due to the degradation of the Reactor Protection System and vital instrumentation. The first phase requires re-energization of the A.C. vital panel within two hours. During this phase the panel may be powered by a Class 1E constant voltage transformer (CVT) fed from a vital MCC. However, the condition is permissible for only 24 hours as the second phase of the ACTION requires re-energization of the A.C. vital panel from an inverter connected to its associated D.C. bus within 24 hours. *Alternatively, completion times can be determined in accordance with the Risk Informed Completion Time Program.* Failure to satisfy these ACTIONS results in a dual unit shutdown.

TS 3.8.3.1, Action d

With two or more A.C. vital panels either not energized from their associated inverters, or with the inverters not connected to their associated D.C. buses, the required Action is to reenergize the A.C. vital panels from their associated inverters connected to their associated D.C. buses within one hour to regain AC electrical power to the vital buses. The one-hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one required inverter. Alternatively, a completion time can be determined in accordance with the Risk Informed Completion Time Program.

TS 3.8.3.1, Action d is modified by a Note stating it is not applicable when two or more A.C. vital panels are intentionally either not energized from their associated inverters, or the inverters are not connected to their associated D.C. buses. This Action is not intended for voluntary removal of redundant systems or components from service. The Action is only applicable if one bus is not energized from its associated inverter or the inverter is not connected to its associated DC bus for any reason and additional buses are found not energized from their associated inverter or the inverters are not connected to their associated DC bus, or if two or more buses not energized from their associated inverter or the inverters are not connected to their associated DC bus are found at the same time.

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3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

Therefore, the correct interpretation of the footnote for the swing LCs and MCCs is as follows:

Electrical bus can be energized from either train of its unit (establishes the associated bus) and swing function to opposite train must be OPERABLE for the Units in MODES 1, 2, 3, and 4 (or the opposite train is INOPERABLE).

The swing load centers are used to supply shared system and cross-unit loads, and other Technical Specification ACTION statements may be invoked for loss of swing capability. As discussed above, the Unit 3 DC battery chargers 3A2 and 3B2 are powered from Unit 4 via swing MCC 4D, and the Unit 4 DC battery chargers 4A2 and 4B2 are powered from Unit 3 via swing MCC 3D. Inoperability of the swing capability could impact both units if any of the swing battery chargers is credited for satisfying Technical Specification 3.8.2.1. Both EDGs are required to be OPERABLE for a swing battery charger. An inoperable swing function prevents one EDG from supporting that battery charger, and a dual-unit 72 hour ACTION statement *or a completion time determined in accordance with the Risk Informed Completion Time Program* applies in accordance with TS 3.8.2.1 ACTION statement a.

With a unit shutdown one 4160-volt bus on the associated unit can be deenergized for periodic refueling outage maintenance. The associated 480-volt Load Centers can then be cross-tied upon issuance of an engineering evaluation.

For the shutdown unit, the swing load center does **NOT** have to be powered from a diesel backed source, since:

a) Technical Specification 3.8.3.2 only requires that the swing load center be energized. **NO** operability requirements are specified for the swing function (as opposed to the requirements for an operating unit) and

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3/4.8.1, 3/4.8.2 & 3/4.8.3 (Continued)

b) The only accident postulated to occur in Modes 5 and 6 is a fuel handling accident. Loss of offsite power is **NOT** assumed to occur concurrently with these events. Additionally, there is **NO** causal relationship between a fuel handling event and a loss of offsite power. Thus, from a design basis standpoint, all of the Control Room HVAC safety functions can be accomplished with the swing load center energized from an offsite source.

Operating units on the other hand are subject to accidents that can both affect the grid, and release radioactivity to the outside environment, e.g., LOCA, MSLB. Thus, to satisfy the design basis requirements for the Control Room HVAC system when a unit is in MODES 1 - 4, the swing load center must be powered from a diesel-backed source.

For an operating unit, the swing load center also has to be powered from a diesel-backed source to be considered OPERABLE. The swing load center is considered to be powered from a diesel-backed source if:

a) It is connected to an electrical power train that has an OPERABLE diesel generator, or

b) It can automatically transfer to a bus that has an OPERABLE diesel generator.

If Load Center H is energized from a load center (either C or D) that does **NOT** have an OPERABLE Emergency Diesel Generator aligned to it and the swing function is also inoperable, then a 2-hour or a 72 hour LCO *or a completion time determined in accordance with the Risk Informed Completion Time Program* would have to be entered, depending on the battery charger requirements (Technical specification Tables 3.8-1 and 3.8-2).

The swing load center will momentarily de-energize any time it transfers between supply busses (manual, automatic, or test conditions). Since this is the specified manner of operation, the momentary load center de-energization does **NOT** require entry into the Technical Specification 3/4.8.3.2 action statement.

Turkey Point Nuclear Plant, Units 3 and 4

Enclosure 1

LIST OF REVISED REQUIRED ACTIONS TO CORRESPONDING PRA FUNCTIONS

PURPOSE

Section 4.0, Item 2 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) identifies the following license amendment request (LAR) content needed on applicable Technical Specifications (TS), comparison of the TS functions to the probabilistic risk assessment (PRA) functions, and comparison of design basis assumptions to the scope of the PRA:

- The LAR will provide identification of the TS Limiting Conditions for Operation (LCO) 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 emergency core cooling system (ECCS) flow rates) for each of the TS requirements, or an appropriate disposition or programmatic restriction will be provided.

SCOPE

This enclosure provides confirmation that the Turkey Point (PTN) 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 Item 2 of the NRC 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, as identified in Table 1 of the LAR.

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 TS with the associated safety analyses, the analogous PRA functions, and the results of the comparison:

- Column "TS LCO/Condition": Lists all of the LCOs and condition statements within the scope of the 4B implementation.
- 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 function(s) 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 "Disposition": 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.

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 (for example, SC-B1) 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.

The calculated RICT is 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 3 model calculation. Due to the close similarity between the Unit 3 and Unit 4 models, the Unit 3 RICTs are considered adequate estimates for the Unit 4 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.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
3.3.1 Reactor Trip System Instrumentation Function 1 – Manual Reactor Trip	2 channels	No	(1) Manually trip reactor on demand	(1) 1 of 2 channels	(1) Not modeled - see Disposition	The operator action for failure to actuate a manual reactor trip will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
3.3.1 Reactor Trip System Instrumentation Function 19 – Reactor Trip Breakers (RTB)	2 RTBs	Yes	(1) Open RTB for an automatic or manual reactor trip signal	(1) 1 of 2 RTBs	(1) SAME	The RTB may conservatively be assumed non-functional when either diverse trip feature is inoperable to conservatively bound the risk increase as permitted by NEI 06-09. The success criteria in the PRA are consistent with the design basis criteria.
3.3.2 Engineered Safety Features Actuation System (ESFAS) Instrumentation			(1) Associated ESF action will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint			Only a limited subset of functions associated with LCO 3.3.2 are in the proposed scope of the amendment request, as discussed below for each function.
Function 1a – Safety Injection (SI) – Manual Initiation	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	The operator action for failure to actuate a manual SI will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
Function 1c – Safety Injection (SI) - Containment Pressure - High	3 Channels	Yes		(1) 2 of 3 channels	(1) SAME	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.
Function 1d – Safety Injection (SI) – Pressurizer Pressure - Low	3 channels	Yes		(1) 2 of 3 channels	(1) SAME	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.
Function 1e – Safety Injection (SI) – High Differential Pressure Between the Steam Line Header and any Steam Line	3 channels per steamline	Yes		(1) 2 of 3 channels in any steamline	(1) SAME	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.
Function 1f – Safety Injection (SI) – Steam Line flow - High Coincident with: Steam Generator (SG) Pressure - Low or T_{avg} - Low	2 steam line flow channels per steam line 1 SG pressure channel per SG 1 T_{avg} channel per loop	Yes		(1) 2 of 3 SGs	(1) SAME	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.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
Function 2b – Containment Spray – Containment Pressure High-High Coincident with : Containment Pressure - High	3 Channels	Yes		2 of 3 channels	(1) SAME	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.
Function 3a1 – Phase A Isolation – Manual Initiation	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	SSCs for the manual Phase A isolation can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a large, pre-existing containment leak; this is bounding for risk associated with an inoperable manual Phase A Isolation channel, and can be used as a bounding surrogate.
Function 3b1 – Phase B Isolation – Manual Initiation	2 pushbuttons	No		(1) 2 of 2 pushbuttons	(1) Not modeled - see Disposition	SSCs for the manual Phase B isolation can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a large, pre-existing containment leak; this is bounding for risk associated with an inoperable manual Phase B Isolation channel, and can be used as a bounding surrogate.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
Function 6b – Auxiliary Feedwater (AFW) – Steam Generator (SG) Water Level – Low-Low	3 channels per SG	No		(1) 2 of 3 channels on any SG	(1) Not modeled - see Disposition	SSCs for AFW actuation on SG Water Level – low-low can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a common cause miscalibration error of all level sensors; this is bounding for risk associated with inoperable channel(s), and can be used as a bounding surrogate.
Function 7a – Loss of Power – 4.16 kV Busses A and B (Loss of Voltage) and Function 7b – Loss of Power – 480 V Load Centers Undervoltage Coincident with SI and Function 7c – Loss of Power – 480 V Load Centers Degraded Voltage	2 channels per bus or load center	No		(1) 2 of 2 channels	(1) Not modeled - see Disposition	SSCs for Loss of Power on 4.16 kV busses or load centers can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a common cause miscalibration error of undervoltage sensors; this is bounding for risk associated with inoperable channel(s), and can be used as a bounding surrogate.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
3.4.2.2 Reactor Coolant System (RCS) - Operating	3 code safety valves	Yes	(1) Prevent RCS pressure from exceeding the safety limit pressure	(1) 3 of 3 code safety valves (for limiting transient)	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The design basis event is a loss of load with immediate reactor trip on loss of load not credited, and operation of pressurizer power-operated relief valves and steam dump valves not credited. Non-ATWS events with partial RPS failure are not probabilistically significant, so they are not considered in the PRA. The success criteria in the PRA for the limiting ATWS events are consistent with the design basis criteria.
3.5.1 Accumulators	3 Accumulators	Yes	(1) Initial cooling mechanism during large RCS pipe ruptures	(1) 2 of 3 accumulators to intact cold legs	(1) SAME	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.

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

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	Disposition
3.5.2 Emergency Core Cooling System (ECCS) Subsystems – T_{avg} Greater Than or Equal to 350°F	4 safety injection (SI) pumps 2 Residual Heat Removal (RHR) heat exchangers 2 RHR pumps Flowpaths from refueling water storage tank (RWST)and containment sump	Yes	(1) Sufficient emergency core cooling in the event of a LOCA (2) Long term core cooling in the recirculation mode	(1) 2 of 4 SI and 1 of 2 RHR pumps (2) 1 of 2 RHR pumps with suction from containment sump supplying suction of 1 of 4 SI pumps; flowpaths to cold legs and hot legs	(1) 1 of 4 SI and 1 of 2 RHR pumps (2) SAME, except • Hot leg injection required only for large LOCA • Only 1 injection line required small or medium LOCAs • Alternate long term heat removal strategies credited for small LOCAs (RCS depressurization and RHR cooling, RWST replenishment, cross-tie of opposite unit RWST	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The PRA success criteria differ from the design basis in 1) not requiring hot leg injection except for large LOCAs, 2) not requiring injection into a both intact RCS loops for small and medium LOCAs 3) not requiring recirculation for small LOCAs if alternate heat removal is available, and 4) only requiring one SI pump instead of two. Success criteria in PRA are based on plant-specific realistic analyses consistent with the PRA standards for capability category II.
3.5.4 RWST	1 RWST per unit	Yes	(1) Sufficient water for ECCS injection for a LOCA to permit recirculation (2) Reactor will remain subcritical following a LOCA with all control rods out of core	(1 and 2) Refueling Water Tank boron concentration, temperature, and level within limits	(1 and 2) SAME	The PRA does not explicitly model the impact of out of limit boron or temperature, but conservatively these can be addressed for the RICT Program by assuming the RWST is unavailable. Therefore, this LCO condition can be evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.

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

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	Disposition
3.6.1.3 Containment Air Locks	2 air locks	No	(1) Meet restrictions on containment integrity and containment leak rate	(1) 2 of 2 airlocks with both doors closed and within leakage limits	(1) Not modeled – see Disposition	SSCs for the containment air locks can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a large, pre-existing containment leak; this is bounding for risk associated with an inoperable air lock door with at least one door closed, and can be used as a bounding surrogate.
3.6.1.7 Containment Ventilation System	Purge Supply and Exhaust Isolation Valves	No	(1) Each valve sealed closed unless open for specific purposes	(1) Each valve sealed closed or open as permitted	(1) Not modeled – see Disposition	SSCs for the containment purge supply and exhaust isolation valves can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a large, pre-existing containment leak; this is bounding for risk associated with an inoperable air lock door with at least one door closed, and can be used as a bounding surrogate.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
3.6.2.1 Containment Spray (CS) System	2 CS trains	Yes	(1) Depressurize containment following a LOCA	(1) 1 of 2 CS trains	(1) SAME	SSCs are modeled consistent with the TS scope. (Failure of CS function does not directly impact either core damage or large early release mitigation, but is modeled for level two PRA.) The success criteria in the PRA are consistent with the design basis criteria.
3.6.2.2 Emergency Containment Cooling System	3 cooling units	Yes	(1) Containment heat removal	(1) 1 of 3 cooling units	(1) SAME	SSCs are modeled consistent with the TS scope. (Failure of the Emergency Containment Cooling function does not directly impact either core damage or large early release mitigation, but is modeled for level two PRA.) The success criteria in the PRA are consistent with the design basis criteria.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
3.6.4 Containment Isolation Valves	2 isolation valves on each penetration	Yes (in part)	(1) Each containment penetration isolated within the time limits assumed in the LOCA analyses	(1) 1 of 2 isolation valves per penetration isolate within required time.	(1) SAME for PRA modeled penetrations. All other penetrations evaluated as not significant sources of fission product leakage and are screened out.	SSCs for containment isolation valves not in the PRA model can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a large, pre-existing containment leak; this is bounding on risk on an inoperable isolation valve and can be used as a bounding surrogate. The PRA does not explicitly model the impact of excessive isolation time. This condition can be addressed for the RICT Program by conservatively assuming the inoperable containment isolation valve is unclosable if it is open. Otherwise, the success criteria in the PRA are consistent with the design basis criteria.
3.7.1.2 Auxiliary Feedwater (AFW) System	3 pumps	Yes	(1) Supply feedwater to SGs to reduce RCS temperature to less than 350°F in the event of a loss of offsite power	(1) 1 of 3 pumps	(1) SAME	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.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
3.7.1.3 Condensate Storage Tanks (CST) System	2 CSTs (shared with both units)	Yes	(1) Source of water to SGs for removing heat from RCS	(1) 1 or 2 CSTs with minimum water volume	(1) SAME	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.
3.7.1.5 Main Steam Isolation Valves (MSIVs)	3 MSIVs	Yes	(1) Ensure no more than one SG blows down in the event of a steam line rupture	(1) MSIV on affected steamline closes	(1) SAME	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.
3.7.2 Component Cooling Water (CCW) System	3 pumps 2 heat exchangers	Yes	(1) Cooling capacity for continued operation of safety-related equipment during normal and accident conditions	(1) 1 pump and 2 heat exchangers)	(1) SAME	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.
3.7.3 Intake Cooling Water (ICW) System	3 pumps 2 headers	Yes	(1) Cooling capacity for continued operation of safety-related equipment during normal and accident conditions	(1) 1 pump and header	(1) SAME	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.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
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**	**Disposition**
3.8.1.1 AC Sources - Operating	2 startup transformers 4 diesel generators (DG)	Yes	(1) Sufficient power for safe shutdown and mitigation and control of accident conditions	(1) Automatically power associated ESF busses	(1) SAME	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.
3.8.2.1 DC Sources - Operating	4 battery banks and associated chargers and motor control centers (MCC)	Yes	(1) Sufficient power for safe shutdown and mitigation and control of accident conditions	(1) Aligned to provide power to busses	(1) SAME	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.
3.8.3.1 Onsite Power Distribution - Operating	3 trains of AC busses 8 pairs of120V vital panels 4 DC busses	Yes	(1) Sufficient power for safe shutdown and mitigation and control of accident conditions	(1) Aligned to provide power to associated busses as specified	(1) SAME	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.

Table E1-2: Unit 3 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate (Days)[1]**
3.3.1 Reactor Trip System Instrumentation	
Action 1 One of two manual reactor trip channels inoperable	30 [3]
Action 1a Two of two manual reactor trip channels inoperable	30 [3]
3.3.1 Reactor Trip System Instrumentation	
Action 10 One of two RTBs with one diverse trip feature inoperable	30 [4]
Action 10a Two of two RTBs with one diverse trip feature inoperable	0.56 [4]
3.3.2 ESFAS Instrumentation	
Action 28 One of two manual SI channels inoperable	30 [5]
Action 29 Two of two manual SI channels inoperable	30 [5]
3.3.2 ESFAS Instrumentation	
Action 26 One of three containment pressure SI channels inoperable	30
Action 27 Two of three containment pressure SI channels inoperable	30
3.3.2 ESFAS Instrumentation	
Action 26 One of three low pressurizer pressure SI channels inoperable	30
Action 27 Two of three low pressurizer pressure SI channels inoperable	30
3.3.2 ESFAS Instrumentation	
Action 26 One of three differential pressure SI channels in one SG inoperable	30
Action 27 Two of three differential pressure SI channels in one SG inoperable	30
3.3.2 ESFAS Instrumentation	
Action 25 One low T_{avg} SI channel in one loop inoperable	30
3.3.2 ESFAS Instrumentation	

Table E1-2: Unit 3 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate (Days)[1]**
Action 26 One of two steam line flow SI channels in one SG inoperable	30
Action 27 One of two steam line flow SI channels in two SGs inoperable	30
3.3.2 ESFAS Instrumentation	
Action 26 One SG pressure SI channel in one SG inoperable	30
Action 27 One SG pressure SI channels in two SGs inoperable	30
3.3.2 ESFAS Instrumentation	
Action 26 One of two containment pressure CSAS channel inoperable	30
Action 27 Two of three containment pressure channels inoperable	30
3.3.2 ESFAS Instrumentation	
Action 28 One of two manual containment isolation phase A or phase B channels inoperable	30 [7]
Action 29 Two of two manual containment isolation phase A or phase B channels inoperable	30 [7]
3.3.2 ESFAS Instrumentation	
Action 26 One SG water level channel in one SG inoperable	30 [8]
Action 27 Two SG water level channels inoperable	30 [8]
3.3.2 ESFAS Instrumentation	
Action 18 One loss of voltage or degraded voltage sensor on one bus inoperable	18.88 [8]
Action 18a Two loss of voltage or degraded voltage sensors	9.33 [8]
3.4.2.2 RCS - Operating	
Action (undesignated) One of three code safety valves inoperable	30
3.5.1 Accumulators	

Table E1-2: Unit 3 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate (Days)[1]**
Actions a and b One accumulator inoperable	30
Action c Two accumulators inoperable	30
3.5.2 ECCS Subsystems - Operating	
Action a One ECCS component or flow path (other than SI or RHR pump) inoperable	6.24
Action d Two SI pumps inoperable, both units in MODEs 1 - 3	3
Action e One SI pump inoperable, opposite unit in MODEs 4 - 6	3
Action f One SI pump not capable of being powered from its associated diesel generator	30
Action g One RHR pump inoperable	7
Action h Two or more ECCS components or flow paths (other than SI or RHR pump) inoperable	9.78
Action i Three or more SI pumps inoperable	0.99
Action j Two or more SI pumps inoperable and opposite unit in Mode 4, 5, or 6	0.62
Action k Two or more SI pumps not capable of being powered from their associated diesel generator	0.99
Action l Two RHR pumps inoperable	1.32
3.5.4 Refueling Water Storage Tank	
Action (undesignated) RWST inoperable	8.54 [6]
3.6.1.3 Containment Air Locks	
Action b One Containment Air Lock inoperable	30 [2, 7]
3.6.1.7 Containment Ventilation System	
Action a and b Containment purge supply and/or exhaust valve(s) exceeding leakage rate	30 [2, 7]

Table E1-2: Unit 3 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate (Days)[1]**
3.6.2.1 CS System	
Action a One CS train inoperable	30 [9]
Action b Two CS trains inoperable	30 [9]
3.6.2.2 Emergency Containment Cooling System	
Action a One unit inoperable	30 [9]
Action b Two or more units inoperable	30 [9]
3.6.4 Containment Isolation Valves	
Actions a, b, or c One or more containment isolation valves inoperable	30 [2, 7]
3.7.1.2 AFW	
Action 1 One train inoperable	30
Action 2 Two trains inoperable	0.81
3.7.1.3 Condensate Storage Tank	
Action undesignated Condensate Storage Tanks with less than 210000 gallons - opposite unit in MODEs 4 - 6	14.65 [6]
Action 1 Condensate Storage Tanks with less than 420000 gallons but greater than 210000 gallons - both units in MODEs 1 - 3	23.57 [6]
Action 2 Condensate Storage Tanks with less than 210000 gallons - both units in MODEs 1 - 3	0.67 [6]
3.7.1.5 Main Steam Isolation Valves	
Action a One MSIV inoperable	30
Action b Two or more MSIVs inoperable	30
3.7.2 CCW System	
Action b One OPERABLE pump, or two OPERABLE pumps without	13.54

Table E1-2: Unit 3 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate (Days)[1]**
independent power supplies	
Action c Less than two heat exchangers OPERABLE	0.63
Action d Three CCW pumps inoperable	0.60
3.7.3 ICW System	
Action a Only two pumps OPERABLE with independent power supplies	30
Action b One OPERABLE pump, or two OPERABLE pumps without independent power supplies	16.08
Action c One header OPERABLE	30
Action d Three ICW pumps inoperable	0.56
Action e Two ICW headers inoperable	0.53
3.8.1.1 AC Sources - Operating	
Action a One of two startup transformers inoperable	29.72
Action b One required diesel generator inoperable	30
Action b** One required diesel generator inoperable and one startup transformer inoperable	3
Action e Two of two startup transformers inoperable	27.81
Action f Two diesel generators inoperable	9.95
Action I Three or more AC sources inoperable	5.65
3.8.2.1 DC Sources - Operating	
Action a One battery charger not capable of being powered from its associated diesel generator	30
Action b One battery bank inoperable	29.51
Action b* One battery bank inoperable and opposite unit in mode 5 or 6	9.33

Table E1-2: Unit 3 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate (Days)[1]**
Action c Two or more battery banks inoperable	5.91
3.8.3.1 Onsite Power Distribution - Operating	
Action a One train of AC emergency busses not fully energized	0.33
Action c One AC vital panel not energized from its associated inverter or connected to its associated DC bus	4.96
Action d Two AC vital panels not energized from their associated inverter or connected to its associated DC bus	4.62
Action e One DC BUS not energized from its associated battery bank or charger	2.89

1 Actual results are provided, but RICT is limited to a maximum of 30 days.
2 LERF is the limiting risk metric.
3 Model fails manual actuation of reactor trip.
4 Model fails one RTB when either diverse trip feature is inoperable.
5 Model fails manual actuation of the associated ESF function.
6 RICT evaluated for limiting condition of the RWST being empty.
7 RICT evaluated for limiting condition of loss of containment function for a large containment penetration.
8 Model fails all actuation channels of the associated ESF function.
9 The system is modeled for the level 2 PRA, but has no impact on core damage or large early release.

6.0 REFERENCES

1. ML071200238, *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 (TAC No. MD4995)*, Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.
2. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document*, Nuclear Energy Institute, Revision 0, November 2006.
3. PTN-BFJR-00-001, Revision 10, PTN PSA Model Update – EPU, 7/16/12.

Turkey Point Nuclear Plant, Units 3 and 4

Enclosure 2

INFORMATION SUPPORTING CONSISTENCY WITH REGULATORY GUIDE 1.200, REVISION 2

Introduction

NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 1) Section 2.3.4 identifies that the probabilistic risk assessment (PRA) shall be reviewed using the guidance of Regulatory Guide (RG) 1.200 (Reference 2) for a PRA which meets Capability Category II for the supporting requirements (SRs) of the internal events at power PRA standard (Reference 3), and that deviations shall be justified and documented. Section 4.0, Item 3 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 4) for NEI 06-09 requires the license amendment request (LAR) to include a discussion of the results of peer reviews and self-assessments conducted for the plant-specific PRA models which support the RMTS Program, including the resolution or disposition of any identified deficiencies (i.e., findings and observations from peer reviews). The scope of this information includes the internal events PRA model, and other models for which additional standards have been endorsed by a revision to RG 1.200.

This enclosure provides information on the technical adequacy of the Turkey Point PRA internal event, internal flood, and fire models which support the Risk-Informed Completion Time (RICT) Program, in support of the LAR to revise Technical Specifications (TS) to implement NEI 06-09. This information is consistent with the requirements of Item 3 of Reference 3, and addresses each PRA model for which a RG 1.200 endorsed standard exists. The information is provided as follows:

Table 1 Internal Event and Internal Flooding PRA Model Peer Reviews

Table 2 Internal Fire PRA Model Peer Review

Note that other external hazards including seismic hazards are not addressed by PRA models, and are further discussed in Enclosure 4. Shutdown modes of operation are not in the scope of the RICT Program, and so low-power and shutdown PRA models are not addressed. No other PRA standards are endorsed by RG 1.200.

No changes have been made to the internal event, internal flood, or fire PRA models since the peer reviews that would constitute an upgrade as defined by ASME/ANS RA-Sa-2009, and therefore no additional focused-scope peer reviews are required to support implementation of the RICT Program. Future changes to the Turkey Point PRA models will be performed consistent with station procedures for design changes, procedure changes, and equipment performance monitoring. This will also include updates to implemented risk-informed applications as applicable and appropriate.

Internal Event and Internal Flood PRA

FPL conducted a full-scope PRA peer review (Reference 7) in January 2002 using the NEI 00-02 process. Following the issuance of the ASME PRA Standard and Regulatory Guide 1.200, an internal gap analysis was performed where the findings from the original 2002 peer review

were incorporated into the overall assessment of the PRA's quality with respect to the Standard's supporting requirements. The current Turkey Point gap analysis uses the RA-Sa-2009 version of the standard as endorsed by RG 1.200, Revision 2.

To supplement the original peer review and internal gap analysis, and to further verify the quality of the updated internal events model used in the Fire PRA, in April 2011, a focused peer review (Reference 8) was performed assessing the human reliability analysis (HRA) and internal flooding analysis portions of the PRA using the latest PRA standard, ASME/ANS RA-Sa–2009, and Regulatory Guide 1.200, Rev. 2.

Most recently, a focused peer review (Reference 9) was performed in October 2013, to assess portions of the PRA model which had received upgrades: common-cause failure analysis, Level 2 analysis, and interfacing system LOCAs.

The peer reviews identified findings and observations (F&Os) for SRs of the internal event and internal flood PRA models. The initial 2002 global peer reviews ranked the findings as A, B, C, or D, based on their significance ("A" being the most significant). The focused peer reviews included: findings (F) for elements which did not meet at least Capability Category II of a SR of the standard, suggestions (S) from the peer review team for elements which met the SR but could be improved, and strengths (STR). The F&Os categorized by the peer review team as findings have been resolved by PRA model revision, including documentation updates, with the exception of those from the final, 2013 focused peer review, whose F&Os are unresolved at this time. All of the peer review F&Os are presented in Table 1, Internal Event and Internal Flooding PRA Peer Review – Findings Resolution.

Fire PRA

A peer review of the Turkey Point (PTN) Fire PRA was performed in February 2010 at PTN using the NEI 07-12 Fire PRA peer review process, the combined PRA standard, ASME/ANS RA-Sa–2009, and RG 1.200, Revision 2. The purpose of this review was to provide a method for establishing the technical quality and adequacy of the Fire PRA for the spectrum of potential risk-informed plant licensing applications for which the Fire PRA may be used. The February 2010 PTN Fire PRA Peer Review (Reference 10) was a full-scope review of all the Technical Elements of Part 4 of the ASME/ANS Standard. A subsequent peer review, performed in March 2012, was a focused scope peer review (Reference 11) addressing the FSS, HRA and PRM Technical Elements. The report was finalized and issued to PTN in May 2012.

The Fire PRA update addressed the Supporting-Requirement-assessed deficiencies (i.e., Not Met or CCI). Completion of recommendations related to Supporting Requirement assessments and 'Finding' F&Os resulted in a Capability Category II assessment for the majority of the Supporting Requirements.

The F&Os from these peer reviews are presented in Table 2.

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
AS-1	The following items were observed related to the success criteria: (1) For small - small LOCAs where high pressure recirculation fails, the ECA-1.1 (Loss of ECC recirculation) actions to refill the RWST via the CVCS and continue injection are modeled. If this succeeds, the modeled end state is successful core cooling. Although crediting the action is valid, the sequence as modeled has not necessarily reached a stable end state; additional action in the long term is required to put the plant in a stable state. For example, the RWST refill can be argued to extend the accident sequence mission time past 24 hours and therefore beyond the current Level 1 PRA model scope. If the additional time were long, then in taking credit for these strategies the impact on pump and other component run failures, and any additional actions to achieve a stable state should be modeled. In addition, some evaluation should be included regarding potential effects on containment instrumentation or components of increasing water level. (2) For ATWS sequences where "Reactivity Control Late" is asked, the model credits emergency boration, per procedure FR-S.1. The procedure, and plant training documents, indicate that 1 of 3 charging pumps are needed to ensure at least 60 gpm of borated injection for shutdown. However, the fault tree model (at gate U3WRCL1) implements this as failure if any single charging pump fails. This is incorrect and should be fixed.		B	Closed	(1) This item has been resolved. For the small-small LOCA sequences where RWST is credited, secondary cooling is available, permitting depressurization and reduction of the leak rate, making RWST refill viable. In most of these sequences, some fault(s) in the RHR system is preventing successful recirculation or shutdown cooling. The small flow rate associated with a depressurized small-small LOCA and successful RWST refill and available secondary cooling permit continued decay heat removal and inventory control for an extended period. The concern over-extending the mission time beyond 24 hours is offset by the increasing probability of recovering hardware failures in the cutset which initially prevented initiating recirculation or shutdown cooling. Further, if RWST refill fails, HHSI from the opposite-unit pumps and RWST is available. (2) The charging pump success criterion has been changed to 1/3 charging pumps for emergency boration to match that in the success criteria calculation.	2002 WOG

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
AS-2	Several inconsistencies between the success criteria as stated in the Accident Sequence Analysis Notebook and the linked fault tree model. Specific examples are: a. The Small-small LOCA success criteria for early core heat removal is listed in Table 3 as 2/4HHSI pumps and 1/3 AFW pumps. However, fault tree gate G1PMP3 shows 1 HH pump required for small-small LOCA. b. The Small LOCA success criteria for early core heat removal is shown as 2/4 HHSI pumps OR 1/2 RHR and depressurization. However, section 4.1 of the Accident Sequence Analysis Notebook and fault tree gate U3S2CD2 only credit HHSI. c. The Medium LOCA success criteria for early core heat removal is listed as 2/4HHSI pumps in Section 5.1 and Table 5 of the Accident Sequence Analysis Notebook. However, fault tree gate G1PMP3 and the supporting MAAP analyses from the IPE show only 1 of 4 HH pumps to be required. d. The success criteria for early core heat removal using the AFW system is described differently in the PTN System Analysis Notebook and the Accident Sequence Analysis Notebook. The fault tree modeling appears to be generally consistent with the criteria stated in the System Analysis Notebook. For example, the System Analysis Notebook states that for ATWS, the AFW system must supply flow from 2 AFW pumps through all six AFW control valves. In Table 8 of the Accident Sequence Analysis Notebook, the ATWS success criteria for AFW is stated as 2 AFW pumps to 3/3 SGs. The structure of fault tree gate A0201 agrees with the System Notebook criteria rather than the Accident Sequence Analysis Notebook. Similar differences were noted in the success criteria descriptions related to the heat removal requirements for the Transient and Small-small LOCA sequences where the Accident Sequence Analysis only gives the pump success criteria without including the requirement to provide flow through at least 3 of 6 flow control valves..		B	Closed	This item has been resolved via the update of the Accident Sequence Analysis Notebook, the new success criteria calculation, and model updates.	2002 WOG

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
AS-3	The Accident Sequence Analysis Notebook lists several systems available to provide various success criteria which are not credited in the linked fault tree. For example: a. The Transient success criteria for long term core cooling does not credit RHR Shutdown Cooling, Low-head Recirculation, RWST replenishment with continued HHSI, Depressurization with Low-head injection and low-head recirculation, continued charging with RWST replenishment, or continued HHSI with opposite Unit RWST. b. The Small-small LOCA success criteria do not credit Bleed and Feed and the long-term cooling success criteria does not credit low-head recirculation or continued HHSI using the opposite unit RWST c. The Small LOCA success criteria for early core heat removal does not credit depressurization and Low-head injection. In addition, the long-term cooling function takes no credit for RHR cooling, low-head recirculation or opposite unit RWST.		B	Closed	a. Continued HHSI using the opposite unit RWST is now credited for long-term cooling of transients. RWST replenishment is not credited, based on the judgment that the leak (2 PORVs worth) is rather substantial, and no secondary cooling is available. The remaining options may be viable, but have many of the same hardware dependencies as HHSR, and consequently will make little difference and will not be added. b. As for small-small LOCA, continued HHSI using the opposite unit RWST has been added. The addition of low-head recirculation with depressurization would have little effect due to shared dependencies with HHSR and, therefore, will not be added. Credit for bleed-and-feed cooling for the S1B sequences has been added to the model. c. For small LOCA, continued HHSI using the opposite unit RWST has been added. As for the addition of the other options, the addition of low-head recirculation with depressurization would have little effect due to shared dependencies with HHSR, and credit for depressurization and low-head injection was not modeled due to the reduced time available because of the larger break size (2-6").	2002 WOG
AS-9	The SGTR event tree does not have complete sequence delineation. For the case with failure of isolation and failure of HHSI, cooldown and depressurization to RHR conditions is asked and, if successful, no further delineation is provided in the model. There should be some T/H analysis to show that the plant can be cooled down to RHR entry conditions without HHSI given a SGTR.		B	Closed	The SGTR event tree was revised such that no credit for RCS cooldown and depressurization is given for sequences where SG isolation and HHSI fail.	2002 WOG
DA-1	NUREG/CR-4550 was followed to develop CCF. NUREG/CR-4550 was issued in 1986. The approach addressed in NUREG/CR-4550 may have out of date. NUREG/CR-4780 (or equivalent) systematic approach should be followed.		B	Closed	The Turkey Point CCF model was updated to reflect the alpha-factor approach and data from INEL 94/0064.	2002 WOG

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
DA-2	The test and maintenance probabilities used for individual components are based on actual outage time as collected by the plant. The component outage time was clearly collected over the period of time the plant was in Mode 1,2,3. The fault trees and event trees use several cross-ties from AC power, HHSI, and AFW. In the use of these cross-ties, the opposite unit components have T&M events. The opposite unit may be in Mode 4,5,6 at the time of demand and the desired equipment may have lesser Tech Specs than those assumed for power operation. The T&M event probabilities for the opposite unit components must consider unavailability over the total period of demand, not just during power operation. This can be done at the fault logic level (with house events for OOS) or in the data probabilities. Currently, neither is done. The most important case of this is the DG's. The DG T&M unavailability is about 6E-3 (55 hours per year). If the OOS time for major overhaul were considered, the unavailability would be .03 to .05.		B	Closed	Logic was introduced to the model to change the opposite-unit EDG test and maintenance probability during outage conditions through the use of flags representing the operating mode of the unit. These flags were also used to model the effect of the opposite unit's mode on the different system crossties.	2002 WOG
DA-4	The latest data updating was done in 1995 based on plant specific data from 1990 to 1994, and the generic database developed in 1989.		B	Closed	The data used in the latest model update has plant-specific data derived from plant records from 1992 through 2006, and generic data from the latest sources, primarily NUREG/CR-6928.	2002 WOG
DA-7	The following observations on CCF modeling in CCW and ICW: For the ICW and CCW initiating events, RIF of one pump to run for 1 year is coupled with CCF of the remaining 2 pumps to run for 72 hours, but the events are independent. For ICW initiating event, CCF of the strainers are not included. There is no evidence of CCF caused by environmental factors in the ICW initiating event.		B	Closed	For the first comment, it is appropriate that the CCF for the ICW pumps be modeled this way for the initiating event of loss of ICW. Two pumps are normally running with one in standby. CCF for the strainers was added to the model. Review of PTN history showed no reactor trips due to environmental factors (sea grass). The initiating event fault tree for ICW includes CCF failures of the ICW pumps and strainers.	2002 WOG

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
DE-1	Turkey Point switched their common cause methodology from the Beta Factor Approach to the Alpha Factor Approach. It appears that Turkey Point only models the exact combination of failures that lead to system failure. For example, Gate G121 of the HPSI model covers the scenario where the success criterion is 2 of 4 HPSI pumps are required. However, only common cause failure of 3 of 4 pumps is modeled. In this case common cause failure of 4 of 4 pumps should also be modeled. Likewise, for gate G131, which covers the scenario where the success criterion is 3 of 4 HPSI pumps, only common cause failure of 2 of 4 pumps is modeled. In this case, common cause failure of 3 of 4 pumps and 4 of 4 pumps should also be modeled. Turkey Point also does not model common cause failure combinations less than that required for direct system failure but which could lead to system failure when combined with a single random failure. Even for the CCF level modeled, Turkey Point doesn't model all applicable combinations of CCF failures (i.e. for 2/4, there are six combinations). In addition, Turkey Point does not model common cause failure of the 2 Unit 3 EDGs coupled with failure to align to the unit 4 EDGs.		A	Closed	A complete CCF update addressed these issues.	2002 WOG
DE-2	The Flooding analysis considered component vulnerability to both flooding and spray effects. Multiple screening criteria were employed to eliminate areas from further consideration. The three remaining areas were then analyzed in more detail to determine a CDF from flooding. A CDF of approximately 5E-7 was calculated for flooding and determined to be not significant relative to then overall risk (IPE CDF of 3.7E-4). The current model maintains cutsets of lower CDF than these flooding cutsets. The conclusion that the flooding is not significant is no longer supported.		B	Closed	The internal flooding analysis was completely revised for RG 1.200 compliance.	2002 WOG
HR-3	Human errors for inter-unit cross ties are not accounted for under conditions of outage for the other unit and special initiating events.		B	Closed	Flags representing the status of the opposite-unit (operating or shutdown) were added to the model to account for the effect of the opposite unit's mode on the different system crossties. Operator actions for inter-unit crossties are explicitly modeled.	2002 WOG
HR-4	A systematic method applied to handle the dependencies between pre-initiators. A sensitivity study to generate cutsets with the HEPs to set to values of 1.0 is applied to identify post-initiator HEP dependency.		STR	Closed	NA	2002 WOG

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
IE-2	The disposition of dual unit initiators and dual unit success criteria is not clear. The following observations were made: 1) Loss of grid is called a "dual unit initiator" with a frequency of 0.053. The derivation for 0.053 is dominated by switchyard faults, which the IPE notebook implies would be a single unit initiator. 2) Loss of a DC bus on either unit will require the other unit to shut down, but it is not explained why this is not a dual unit initiator. 3) There are no guidelines for dual unit success criteria. There are several shared systems at PTN. The success criteria for Unit 3 assume complete availability of the Unit 4 systems to mitigate events at Unit 3. There should be an identification of dual-unit initiators and development of associated dual-unit success criteria.		B	Closed	1) The LOOP initiators are now split into 5 different initiators. There are 4 dual-unit initiators: plant-centered, weather-induced, grid-related, and grid blackout; and 1 single-unit initiator. The two units share one switchyard, so it is assumed that switchyard faults cause a dual-unit LOOP. The single-unit LOOP is dominated by unit-specific startup transformer faults which are on the periphery of the switchyard and cause a loss of offsite power to only one unit. 2) This is because the other unit would have to shut down due to Tech Specs. As such, it would not be an immediate reactor trip, but a controlled shutdown. 3) Dual-unit success criteria are discussed fully in the most recent AS Notebook and the success criteria calculation. The effects of the dual-unit initiating events on the opposite-unit systems are modeled.	2002 WOG
IE-3	The initiator for Pressurized Thermal Shock is considered "out of scope" with no justification. PTS is not out of scope for PSA. Discussion with PTN PRA staff indicate that from a licensing perspective, it has been determined to be a resolved and therefore should not be in the PRA. This should be explained, with a probabilistic explanation of why it is a small contributor. The initiator for Reactor Vessel Rupture is considered out of scope with no justification.		B	Closed	Added RV Rupture IE in model update. Research over the last few years by NRC, EPRI, DOE, MRP, and others has shown PTS to be a non-issue for plants like Turkey Point.	2002 WOG
IE-4	PTN has several shared systems. These include HHSI, AFW, and DG. The configuration of these systems may depend on the status of the unit. For the purpose of system sharing, the opposite unit equipment is always assumed to be available in Mode 1 operability. The following observations were made: 1) The T&M unavailability for DG implies about 50 hours a year OOS. This can not include time for annual overhaul, (which is done at shutdown). When the opposite unit is in Mode 6, the DG tech spec is reduced to 1 DG. The PRA does not capture this dependency on plant status. 2) When Unit 4 is in Mode 5 or 6, the HHSI for unit 4 is cooled by CCW on unit 3. The PRA does not capture this. 3) there are no overall guidelines and criteria for treatment of the opposite unit's operability and it's affect on equipment availability.		B	Closed	Logic was introduced to the model to change the opposite-unit EDG test and maintenance probability during outage conditions through the use of flags representing the operating mode of the unit. These flags were also used to model the effect of the opposite unit's mode on the different system crossties.	2002 WOG
IE-7	The level of independent technical review of PRA changes is indeterminate. There are no comment and resolution sheets for the PRA modification process. There is only a signoff sheet on the overall update package (which contains several individual changes).		B	Closed	There is now a comment and resolution section in the model update calculations.	2002 WOG

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
IE-9	Random reactor coolant pump seal failure has not been included. From NUREG/CR-5750, this event can be about 1E-3. The S1 frequency is derived from pipe rupture failures only. It does not include component leakage, RCP seal LOCA, or any other sources of small leakage.		B	Closed	A review of the random seal LOCA events underlying the NUREG data revealed that the events were not applicable due to changes in design, operations, and procedures. S1 LOCA frequency is taken from NUREG/CR-6928, which is not limited to pipe ruptures. RCP seal LOCAs are modeled explicitly.	2002 WOG
L2-1	Hot leg creep rupture was assumed to occur 99.99% of time for all sequences, regardless of the RCS pressure. This is based on interpretation of latest NRC research which implied that natural circulation of core debris would occur at the time and with such efficiency that hot leg would fail prior to vessel failure and steam generator tube rupture. Although hot leg rupture may indeed be likely, the way it is modeled precludes any other high pressure phenomena, including DCH, Induced tube rupture.		B	Closed	In the updated Level 2 analysis, the probability of hot leg creep rupture is 95%, not 99.99%, a substantial difference, making a sensitivity analysis unnecessary.	2002 WOG
MU-2	STD-R-001 has a requirement for signoff by the preparer, an independent reviewer and the RRAG supervisor. The PRA update calculations that were reviewed had all the required signatures. However there were no review notes or discussion of the disposition of review comments in the various calcs examined by the peer reviewers. Further, the peer reviewers found examples of inconsistencies in several signed-off notebooks (e.g., Accident Sequence Notebook included incorrect success criteria for S2 LOCAs), and examples of errors carried through several PRA Update Calc revisions (e.g., CDF cutsets that included single failures in emergency boration pumps for ATWS, which should have required multiple failures).		B	Closed	There is now a comments and resolutions section in the model update and other PRA Group calculations. Details of each change are documented in the change database. Each model update includes a table of the changes implemented for that model update and reasons for those changes. Consistency issues between the Accident Sequence Analysis, the Success Criteria calculation, and the model update calculations have been resolved in RG 1.200-related upgrades of all of these documents. The other issues are treated separately in other F&Os.	2002 WOG
MU-3	Procedures STD-R-001, Rev. 0 (Software Control Procedure) and STD-R-002, Rev. 5 (Update and Maintenance) govern model control. The computer models (e.g., current and previous model files such as *.CAF and *.BE) for the Turkey Point PSA are maintained on a server (g:nis\psa). The server is backed up daily and therefore provides secure storage. Access to files on this server is limited to those with permission and is on a read-only basis. In addition, computer models are stored on a CD and sent to the document control center. A second copy is maintained locally. Model changes are also maintained on this server. This same process is used for the PSA software (e.g., EOOS, CAFTA, FORTE).		STR	Closed	NA	2002 WOG
MU-4	STD-R-002 requires a data update every 5 years. However, it does not appear that Turkey Point has updated the reliability data since 1995 even though the common cause failure data, the human factors data, the initiating event data and the unavailability data was updated in 2000.		B	Closed	Data was updated in the current PRA model using the data documented in the data update calculation.	2002 WOG

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
MU-5	STD-R-002 includes a set of criteria for when to perform a model update. The key criteria are if the change has a significant impact on CDF or risk insights. However, there are no criteria as to what constitutes a significant impact. Given that there is no fixed update period, there is a concern that the number of "minor" changes pending could build up until the combined impact is significant without triggering an update.		B	Closed	Changed procedure to have a maximum interval of five years between model updates. Whether minor changes constitute justification for a model update is determined by the model custodian.	2002 WOG
QU-2	The guidance provided for the quantification process is IPE vintage. Several new codes are being used in the current process which did not exist at the time of the IPE. No guidance procedures currently exist to control key processes involved in model integration and quantification such as: a. Criteria for development of the mutually exclusive events file b. Selection of truncation value c. Quantification on a sequence basis versus quantification of top gate d. Process for breaking circular logic in the single top linked fault tree (i.e., selection of proper gate level for performing the logical break, naming scheme to be used for gates in the new system fault tree, etc. e. Process for development of flag files for the baseline quantification to ensure the quantified configuration represents normal plant operation practices f. Selection of parameters for input to the UNCERT code for uncertainty calculation		B	Closed	Changes to the mutually exclusive event combinations, flag file, circular logic breaks, and recovery rule file are documented in the change database and the model updates. Details of the quantification process are documented in the Quantification Notebook and the model updates. Truncation level is set as low as the hardware and software will allow, or until convergence is achieved. Uncertainty analysis input is documented in the model update calculations.	2002 WOG

Table 1 - Internal Event and Internal Flooding PRA Peer Review – Findings Resolution

FO ID	Finding	SR	Level	Status	Disposition	Source*
QU-3	The quantification of a linked fault tree model involves the proper integration of several files which can affect the results. For example: a. The quantification flag file is used to set logic flag events true or false to represent normal system alignment. At PTN, this flag file is also used to set certain maintenance events false. b. The mutually exclusive file is used to remove cutsets from the results file which contain certain combinations of events representing disallowed maintenance or illogical event combinations (i.e., events for failure to open and spurious opening of the same valve in a single cutset). c. The recovery rule file is used to add recovery events to the cutset results based on the appearance of certain combinations of failure events. At PTN, this process is also used to apply human error factors to the quantification results. Since these files control vital processes during quantification, independent review and thorough documentation is needed to ensure that the quantification results do not exclude valid failure sequences. The current mutually exclusive events file (PTN2KMEE.TXT) was changed as a result of the addition of new T&M events for LC/SWGR HVAC AHUs and Sump Level Indicators. The calculation package includes a description of "add double maintenance events for these basic events to mutually exclusive events." However, no justification for making the events mutually exclusive or specifying the combinations that are mutually exclusive is provided. In addition, the review of the mutually exclusive events file indicates that some complimentary combinations related to AFW pump maintenance may not be included. While this would lead to conservative results due to failure to remove invalid cutsets, the addition of inappropriate mutually exclusive combinations would have the opposite result. Similar errors can be introduced through the recovery file through the inappropriate application of recovery events to sequences which do not represent the conditions assumed in the HRA analysis.		B	Closed	Changes to the mutually exclusive event combinations, flag file, circular logic breaks, and recovery rule file are documented in the change database and the model updates. Details of the quantification process are documented in the Quantification Notebook and the model updates. Truncation level is set as low as the hardware and software will allow, or until convergence is achieved. Uncertainty analysis input is documented in the model update calculations.	2002 WOG
QU-5	Documentation was not available to indicate that PTN has performed qualitative evaluation for causes of uncertainty, such as: a. possible optimistic or conservative success criteria, b. suitability of the reliability data, c. possible modeling uncertainties (asymmetry or other modeling limitations due to the method selected), d. degree of completeness in the selection of initiating events, and e. possible spatial dependencies.		B	Closed	In the PRA model updates, sensitivity analyses are run to show the effect of key modeling assumptions. Parametric uncertainty is addressed in the model updated calculations. Comprehensive uncertainty analysis evaluations are provided in the Uncertainty Analysis Notebook.	2002 WOG

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FO ID	Finding	SR	Level	Status	Disposition	Source*
QU-6	Recovery of offsite power is applied to sequences where offsite power may not recover the lost function. This occurs in two types of circumstances: 1) for non-grid-loss initiators, LOSP (and SBO) can occur due to failure in the AC power distribution system. XROSPi is applied. The recovery probability of XROSPi is based on the NSAC document for restoration of offsite power to nuclear plants. The sequence in question is cause by a failure of a breaker or transformer at the plant. It is not clear that the recovery probability is applicable. 2) for some SBO sequences where all SG heat removal is lost, XROSPi is applied. Although AC power is available, it is not clear that SG Heat Removal is restored. The SG heat removal is provide by 3 TD pumps and 1 diesel driven pump. If these fail, restoration of AC power does not make them operable. For example, there is a cutset at 1.274E-9 which is loss of 4KV 3A with failure of aux transformer. There is a recovery for XROS19 and EHFPXTIE. The correctness and reasonableness of this practice is questioned.		B	Closed	In the current model update, the recovery of offsite power is not credited for cutsets where the recovery of offsite power does not recover mitigating equipment sufficient to avoid core damage.	2002 WOG
QU-8	The sub tier criteria for a grade 3 on this element considers the following to be indicative of a good understanding of the dominant risk contributors: a. The accident sequence results by sequence, sequence types, and total should be reviewed and compared to similar plants to assure reasonableness and to identify any exceptions. b. A detailed description of the Top 10 to 100 accident cutsets (CAFTA or NUPRA) or accident sequences (RISKMAN) should be provided because they are be important in ensuring that the model results are well understood and that modeling assumption impacts are likewise well known. c. The dominant accident sequence groups or functional failure groups should also be discussed. These functional failure groups should be based on a scheme similar to that identified by NEI in NEI 91-04, Appendix B. There is no discussion of results in the calculation packages for updates provided to the review team to indicate that this type of evaluation is done of the quantification results. Also, the calculation packages provide no discussion of how the dominant cutsets or important systems were affected by the changes to the model when compared to the previous revision.		B	Closed	a. A comparison of PTN CDF cutsets to Robinson's CDF cutset was made and is documented in the Quantification Notebook. Where differences in the cutsets occurred, they could be explained by design or data differences. b. A list of the top 50 cutsets is provided in the model updates. c. Initiating event pie charts, system importance charts, and a table listing the individual sequence contributions are included in each model update calculation.	2002 WOG

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FO ID	Finding	SR	Level	Status	Disposition	Source*
ST-1	ISLOCA Analysis: The ISLOCA analysis was performed in 1991 as part of the IPE and has not been updated since. It is currently undergoing an update. All containment penetrations were identified for potential ISLOCA paths. Each penetration was screened or considered for further analysis. Penetration #1 contains the RHR shutdown cooling line. There are 2 normally closed MOV's (751 and 750) in the 14 inch line. An incorrect assumption about the break location led to the screening of this penetration. This assumption would not have occurred if the structural strength of the entire pipe segment was considered. The RHR pipe is designed for 2500 lb., to the discharge of MOV 751. The pipe is then 600 lb. pipe to the RHR suction. The pipe is then 150 lb. pipe back to the RWST. There are no check valves in the line from the RHR MOV 751 to the RWST. If the tenets of NUREG/CR-5744 are followed and the ultimate strength of the pipe is considered, the pipe rupture location will not likely be inside containment. If a realistic analysis is performed, a portion of these events will not result in any pipe rupture at all, causing a back flow from the RCS to the RWST.		A	Closed	The ISLOCA analysis has been completely revised and documented. The ISLOCA in the RHR shutdown cooling suction line is no longer screened out.	2002 WOG
SY-2	The RHR/LHSI model for recirculation assumes that failure of either the RWST level indication or the sump level indication will result in a failure to switchover to recirculation. The model for failure of the sump level indication includes common cause miscalibration but common cause miscalibration of the RWST level indicators is not included in the model and no basis could be found for the exclusion of this failure. The CCW model does not include the relief valve, or the surge tank level instrumentation.		B	Closed	Added common cause miscalibration of U3 RWST level indicators and common cause miscalibration of U4 RWST level indicators" to the model.	2002 WOG

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FO ID	Finding	SR	Level	Status	Disposition	Source*
TH-1	The basis for requiring or dismissing HVAC requirements is poorly supported and inconsistent through the PRA. The following observations were made: 1) The HVAC system notebook and DG system notebook require HVAC for EDG rooms, DC equipment room and 4160 equipment rooms. The analysis is based on design basis calculations from A&E done in 1985-1988. 2) Recent updates use engineering judgment and plant experience from system engineer to dismiss need for HVAC to 4160 and DC power rooms. 3) The current fault trees require HVAC for DC room and EDG room for LOCA events only. 4) GOTHIC or other room heat up calculations have not been done to support room cooling of these rooms. Discussions with plant staff during the Certification indicate the following requirements for HVAC: A) Unit 3 DG does not need HVAC. B) Unit 4 DG need HVAC whenever they operate C) the switchgear room needs ventilation to protect the 480v transformer. The switchgear ventilation system is normally running. Remedial action via opening doors and running an exhaust fan is sufficient to maintain temperatures. The lead time and indication of loss of ventilation is sufficient enough that loss of switchgear room ventilation is not considered an initiating event. D) The inverter and battery charger need room cooling. Remedial action is available with plug-in, portable fans, but recovery time is on the order of 1 hour.		B	Closed	The HVAC analysis has been redone to identify and justify required HVAC dependencies, and the issues identified have been resolved.	2002 WOG
TH-3	It is difficult to determine, from the PSA notebooks, which codes or methods of analysis are used for specific success criteria determination, or why these methods are appropriate. For example, applications of the MAAP code, particularly the IPE-vintage 3b version, may require some justification or check for applicability (e.g., avoiding use of MAAP 3.0b for rapid RCS depressurization scenarios, which typically require capabilities beyond what was available in that particular version of the code). Further, it is difficult to determine the specific analytical bases for specific success criteria used in the model. While the Accident Sequence Notebook includes a summary of success criteria for each event, reference for the bases for the success criteria is to the IPE, which does not provide additional information on this subject.		B	Closed	The updated Accident Sequence Analysis Notebook, the new success criteria calculation, and model updates provide explicit bases and references for the success criteria.	2002 WOG

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FO ID	Finding	SR	Level	Status	Disposition	Source*
TH-4	The LOCA break size definitions for the PRA are based on different criteria than those for most other PRAs. This is acceptable if the underlying analyses provide sufficient basis for the definitions. A series of MAAP 3.0b analyses was performed for the Turkey Point IPE. The available documentation consists of the Accident Sequence Notebook descriptions and success criteria summary and an internal memo from the IPE, which provides a summary listing of the MAAP cases that were run, along with an indication as to whether or not core uncovery/vessel failure occurred. Reviewer note R11 to table TH provides a comparison of the definitions and their bases, with focus on the injection phase, as discerned from this information: From the comparison in note R11, it can be seen that the principal difference in size definitions (aside from the names used) is in the PTN Medium Break category, which is essentially the lower end of the typical Large Break category. Comments on the above are as follows: The available documentation provides the basis for some, but not all, of the size ranges noted above. Information provided in FPL memo NF-90-450 (October 19, 1990) provides sufficient information to serve as a basis for the S1 and S2 ranges and the lower end of the Medium LOCA range. But it does not provide any basis for the upper end of the Medium LOCA / lower end of the Large LOCA size ranges (i.e., the 13.5" break). Available MAAP runs listed in the memo are for breaks up to 10" diameter. Discussions with FPL personnel identified that the 13.5" size cutoff may have been selected by the IPE contractor during the early stages of the IPE, but a specific basis was not located during the review. For the TPN Medium LOCA, i.e., breaks up to 13.5 inches, the PRA assumes that a single train of high head injection can mitigate this class of LOCAs, whereas typical PRAs would instead tend to credit a single train of low head injection for breaks at the upper end of this size range (i.e., above 6"). As noted above, analyses supporting the upper end of the Medium LOCA range with this success criterion were not available during the peer review. MAAP 3.0b analyses were used to support the definition of ECCS requirements for the MLOCA, even at the upper end of the break size range (i.e., 13 inches). In general, MAAP 3.0b is not appropriate for rapid depressurizations as would be occurring for breaks in the MLOCA size range.		B	Closed	The updated Initiating Events Notebook, the updated Accident Sequence Analysis Notebook, the new Success Criteria Calculation, and the model updates provide justification for the LOCA break sizes.	2002 WOG
HR-A2-01	This HR requires identification, through a review of procedures and practices, those calibration activities that if performed incorrectly can have an adverse impact on the automatic initiation of standby safety equipment.. The system notebooks contain a detailed listing of testing and maintenance procedures that were identified for each system, but there is no discussion as to which procedures were determined to have the potential to result in equipment being left in a miscalibrated condition, and which were screened from consideration with the basis for screening.	HR-A2	F	Closed	A conservative approach was used for the pre-initiators which assumed pre-initiators are always possible, and detailed evaluations of procedures were made only for risk-significant items. Screening values were used for the non-risk-significant pre-initiators.	2011 FPR

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FO ID	Finding	SR	Level	Status	Disposition	Source*
HR-B2-01	This SR does not allow screening of activities that could simultaneously have an impact on multiple trains of a redundant system or diverse system. In the HHSI system notebook, the following valves are assumed not to be under maintenance while either unit is at power: MOV-*-¬864A, B; *-864C; *-845A, B, C, D; MOV-878A, B; MOV-* 856A, B; * 847C; * 882. Because these valves have the potential to impact BOTH Units, they cannot be screened in this manner. Based on this assumption, these valves would only be worked on while both Units are shutdown, which is probably not realistic.	HR-B2	F	Closed	This F&O addressed the following valves, which were assumed not to be under maintenance while either unit is at power: MOV-*-¬864A, B; *-864C; *-845A, B, C, D; MOV-878A, B; MOV-* 856A, B; * 847C; * 882. For the 864 valves, the model has a T&M event for each RWST to account for the time the RWST contents are used to fill the refueling canal, which is probably the only time the 864 valves could be maintained. The RWST T&M event should be a palatable substitute The 845 and 882 valves are locked-open manual valves, so no T&M or pre-initiator is needed there. The HHSI recirculation valves 856 and 874C, if closed for maintenance take out their related HHSI pumps. The 856 valves are stroke-tested during the associated unit refueling outages. Evaluated pre-initiators for the 856 valves and added these to the model. The 878A and 878B valves, if closed for maintenance, would prevent opposite-unit SI. Evaluated pre-initiators for the 878 valves and added these to the model. The 856 valves are stroke-tested during the associated unit refueling outages. Evaluated pre-initiators for the 856 valves and added these to the model.	2011 FPR
HR-C2-01	There is no provided documentation of the plant-specific or applicable generic operating experience for equipment left unavailable for response in accident sequences.	HR-C2	F	Closed	In the latest data update, condition reports were reviewed for the time period 1992-2006 for component failures. No failure modes outside the ones already modeled were found.	2011 FPR
HR-D1-01	The human failure event probabilities appear to be evaluated with a systematic process that includes an initial screening value and the identification of risk-significant action for which a detailed analysis through ASEP method is used. Although there appear to be some inconsistencies in the values of the HEF, especially for HEF already existing in previous version of the model. For example, action AHFA0N2BK1 is indicated as a pre-existing action (i.e., not highlighted in Table 3, page 22) with an initial value of 1.10E-3. There is no further discussion of this action (i.e., the action is not indicated in Table 4 at page 27 as one of the action requiring further analysis). Still in Table 5 at page 31 the action has a value of 4.5E-5 (consistently with what is in the model). Another example of inconsistency between the documentation, the HRA Calculator file and the CAFTA model is post-initiator action AHFPAFWTHROT).	HR-D1	F	Closed	A review of the model and documentation was performed and the inconsistencies were properly resolved.	2011 FPR

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FO ID	Finding	SR	Level	Status	Disposition	Source*
HR-D3-01	The pre-initiator HRA does not specifically discuss quality of the written procedures or the quality of the human-machine interface.	HR-D3	F	Closed	For each pre-initiator analyzed in detail in the HRA Calculator, there is an assessment of the quality of the procedures and human interface in Performance Shaping Factors.	2011 FPR
HR-G7-01	This SR outlines the requirements for assessing the degree of dependence between HEPs contained in a single accident sequence or cutset, and accounting for the influence of success or failure in preceding human actions and system performance on the HEP under consideration, including consideration of 1) time required to complete all actions in relation to the time available, 2) factors that could lead to dependence including common instrumentation, procedures, increased stress levels, etc., and 3) availability of resources (e.g. personnel). A very detailed dependency assessment has been performed for Turkey Point. However, one potential area that does not appear to have been fully addressed for HEPs credited during dual-unit initiating events is the availability of Operator resources in the Control Room. All HEPs reviewed had "2 ROs" required under the manpower requirements table. However, based on current minimum Control Room staffing, only 3 total ROs are required to support both Units. Therefore, during a Dual Unit initiating event there would not be 2 ROs available to support both Units response times.	HR-G7	F	Closed	The fatigue rule requires the site to have a 4th RO on shift.	2011 FPR
IFPP-B3-01	This SR requires that an uncertainty assessment be included in the documentation. No uncertainty assessment has been included in the documentation.	IF-IFPP-B3	F	Closed	The documentation of the internal flooding analysis now includes a section on uncertainty analysis.	2011 FPR
IFQU-A1-01	No documentation has been provided which identifies that a review of the accident sequences has been performed. In particular, for pipe break events that are MSLB scenarios that have the potential to impact other plant equipment (e.g. TDAFW pumps), ensure that the MSLB accident sequence followed is appropriate. For Circulating Water expansion joint failures, ensure that loss of Condenser Vacuum sequences are followed, etc.	IF-IFQU-A1	F	Closed	The following was added to the internal flooding analysis documentation, "It should be noted that the accident sequences defined in the internal events model were used to quantify internal flooding scenarios. Each scenario description identifies the existing initiating event to which it is mapped. No new sequences or fault tree models were required." The internal flooding analysis documentation now specifically discusses main steam and feed line breaks as being explicitly addressed.	2011 FPR
IFQU-A5-01	No human failure event discussion is presented in the analysis.	IF-IFQU-A5	F	Closed	There is no human failure discussion because no credit was taken for flood-mitigating operator actions.	2011 FPR
IFQU-A7-01	This SR states: PERFORM internal flood sequence quantification in accordance with the applicable requirements described in paragraph 4.5.8. The internal flooding analysis has been quantified in accordance with internal events quantification requirements; however, supporting documentation should be provided which describes the process.	IF-IFQU-A7	F	Closed	The quantification is discussed now in the documentation of the internal flooding quantification.	2011 FPR

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FO ID	Finding	SR	Level	Status	Disposition	Source*
IFSN-A11-01	This SR states: For multi-unit sites with shared systems or structures, INCLUDE multi-unit scenarios. No multi-unit impacts have been quantified or discussed. Multi-Unit shared systems (e.g. TDAFW pumps and HHSI pumps) need to be addressed.	IF-IFSN-A11	F	Closed	Dual-unit impacts are discussed where applicable in individual flood scenarios. The effects of these are automatically accounted for because the Unit 3 and Unit 4 models are linked.	2011 FPR
IFSN-A16-01	This SR provides the criteria under which human mitigative actions can be credited. No discussion of human mitigative actions is provided in the flooding study. It appears that in some cases human mitigative actions are credited, however even this is not clearly stated. See for example the following excerpt from 4.2.1.88 "A 2000-gpm major flood rupture will cause flooding of the Unit 3 and Unit 4 RHR pump and heat exchanger rooms and the submergence damage to the RHR pumps in 2 hours." It is unclear as to why the flood only persists for 2 hours.	IF-IFSN-A16	F	Closed	No credit was taken for flood-mitigating operator actions. The times at which various equipment fail in each scenario do not imply an end to the scenario.	2011 FPR
IFSN-A2-01	No identification of flood alarms or floor drains has been made in the flood analysis document.	IF-IFSN-A2	F	Closed	No credit taken for operator action to mitigate flood; therefore, there was no need to credit flood alarms. Documentation was updated to reflect the fact that drain lines were not credited in determining the impact of a flood in a particular room.	2011 FPR
IFSN-A3-01	This SR states: for each defined flood area and each flood source, IDENTIFY those automatic or operator responses that have the ability to terminate or contain the flood propagation. No justification for how/when Operators determine a flood is in progress, and what triggers them to attempt to isolate it can be found. The front part of the document states that all floods are assumed to last 12 to 24 hours, but a review of the scenarios shows that some are terminated in 2 hours, others in 3 hours, etc. The basis for these times cannot be found. No operator actions regarding termination have been addressed in the analysis. Section 3.1.2 alludes to the fact that floods were generally allowed to persist for 12-24 hours, however discussion found in Section 4.2 seems to contradict this. No information has been provided detailing when floods have been isolated if credit was taken for such an action.	IF-IFSN-A3	F	Closed	No credit was taken for flood-mitigating operator actions.	2011 FPR
IFSN-A4-01	No supporting information has been provided to justify the estimations regarding flood volumes and the subsequent flooding height.	IF-IFSN-A4	F	Closed	The flooding calculations have been added to the internal flooding analysis documentation.	2011 FPR

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FO ID	Finding	SR	Level	Status	Disposition	Source*
IFSN-A6-01	This SR States: For the SSCs identified in IF-C2c, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. EITHER: a) ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions; OR b) NOTE that these mechanisms are not included in the scope of the evaluation. No discussion has been provided for the impact due to the additional flood failure mechanisms.	IF-IFSN-A6	F	Closed	It is now documented in the internal flooding analysis documentation that spray and submergence damage were included in the scope of the evaluation.	2011 FPR
IFSN-A8-01	This SR states: IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via back flow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads. Although the obvious propagation pathways (e.g. doors, stairwells, grating) were identified, a good discussion associated with less obvious pathways (e.g. failed backflow check valves, cable penetrations, cable trays, etc.) for individual zones was not found.	IF-IFSN-A8	F	Closed	Inter-area propagation is discussed in Appendix B of the internal flooding analysis documentation.	2011 FPR
IFSO-A1-01	Based on a confirmatory walkdown performed the Peer Review Team, the locations/impacts of some pipes containing water may have been overlooked in the analysis.	IF-IFSO-A1	F	Closed	The findings involved the chilled water system. The chilled water system operates at very low pressure and the lines are insulated, precluding the possibility of a spray. This information was added to the scenario descriptions.	2011 FPR
IFSO-A3-01	No process by which screening was performed is present in the analysis.	IF-IFSO-A3	F	Closed	No screening was performed; all areas were considered.	2011 FPR
IFSO-A4-01	No human-induced mechanisms have been included in the analysis, and additionally, no process which justifies their exclusion was provided.	IF-IFSO-A4	F	Closed	Human-induced mechanisms are already taken into account in the general failure data.	2011 FPR
IFSO-A5-01	No summary or characterization of flood sources included in the analysis has been provided. It is difficult to tell what the decisions making up the source characterization were.	IF-IFSO-A5	F	Closed	Flooding calculations and discussion of flood sources has been added to the documentation of the internal flooding analysis.	2011 FPR

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DA-D5-01	For several CCF groups, a "global common cause event" (as described at the end of Section 4.2 of PTN-BFJR-2008-012, Rev. 0) is used. While this is a reasonable simplification, the global common cause event needs to account for the common cause combinations that are not included explicitly. However, for several 6-component groups (AFW AOVs FTO, AFW CVs FTO, AFW MOVs FTO), the 5-of-6 term was not included and the 6-of-6 term was not adjusted. A similar issue appears to be present for SG SVs FTO (4-component group), where only the 4-of-4 term is included (the 2-of-4 and 3-of-4 terms are missing and the 4-of-4 term was not adjusted).	DA-D5	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR
DA-D6-01	The CCF notebook did not include a review of plant failure data for common cause events.	DA-D6	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR
DA-D6-02	Section 3.0 of the CCF Notebook includes the assumption that CCFs are not included in fault tree initiating events with year-long mission times due to excessive conservatism in applying CCF factors that are developed for 24-hr mission time. However, this is not sufficient basis for excluding CCFs for fault tree IE models.	DA-D6	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR
IE-C14-01	RCP TBHX rupture probability - The IE frequency for tube rupture is based on a Reference 5 value of 3.48E-08/hr (peer review did not verify this reference) for "HX Tube External Leak Large >50 gpm". This hourly frequency is multiplied by 8760hr/yr for an annual IE frequency of 3.05E-04/yr. Depending on the application of the data, this IE frequency could be applied at each RCP. thus event tree top event "RCP TBHX Tubes Intact?" would be multiplied by a factor of 3. Applicability of the TBHX data to one or all RCPs should be examined/documented for impact on the total %ZZISLTBCCW initiator/results.	IE-C14	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR
IE-C14-02	Manual operator action is credited for local manual closure of MOV-*-626 (should it fail to close) and/or to local closure of manual valve *-736. Operator success ensures that the CCW piping remains intact. Although the HEP for the local action is 0.5, the time window basis should document to ensure that the operator has sufficient time to perform these actions before the CCW piping boundary fails.	IE-C14	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR

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IE-C14-03	Thermal Barrier ISLOCA IE Frequency – RCP Thermal Barrier CCW Supply Penetration #3 - This penetration is not evaluated for potential ISLOCA contribution. This penetration is protected by two normally open, active check valves (717 and 721A/B/C) inside containment and two normally open MOVs (716A/B) outside containment. The associated piping inside containment appears to be designed for full RCS pressure. However, given a thermal barrier tube breach, the active check valves could fail to close (w/CCF). The active failure of the outboard MOVs (also w/CCF) may be highly unreliable due to low differential pressure design capability and lack of relevant closure signals, and there might not be sufficient time for manual action. Failure of this penetration should be assessed for possible contribution to the TBCCW ISLOCA event frequency and sequences.	IE-C14	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR
IE-C14-04	ISLOCA assessment of Penetration 1 (RHR SDC suction line) did not consider that the common suction piping beyond the RHR pumps could be affected by the over-pressurization event. This would impact the function of the high head SI pumps and the RWST (and Containment Spray pumps, which are not important in ISLOCA scenarios). As a result, the current RHR small ISLOCA event sequences apply too much credit for the associated Unit's RWST and HHSI pumps.	IE-C14	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR
IE-C14-05	Penetrations 58/59/60: (HHSI cold leg injection) - These penetrations are qualitatively screened from further detailed evaluation on the basis that ...."the combination of three check valves is equivalent to three locked/closed isolation valves", for meeting NUREG/CR-5928 criterion (c), systems isolated by redundant normally closed and locked manual valves that are independently verified to be closed and locked before plant startup". This comment is also applicable to Penetration 18. Additional basis is needed to support this equivalency assertion for screening these penetrations.	IE-C14	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR
LE-D2-01	Electrical penetration assembly failure modes have been found to be important contributors to overall containment fragility at other large dry PWRs, and in at least 2 instances, tend to be the most limiting in terms of ultimate failure pressure. Additionally, early studies at Sandia National Laboratories have considered the potential impact of very high (beyond design basis) temperatures on elastomer seals (this latter issue is more critical for small volume containments such as BWR Mark I).	LE-D2	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR
LE-F1-01	End state frequency totals are given in Table 5 of the Level 2 notebook, PTN-BJFR-99-010, Rev. 1, and results by release category are given in Table 6. However, results using the Plant Damage State definitions of Section 4.2 are not provided. CC II is not met because relative contribution to LERF by PDS is not shown, although information is available to provide such data.	LE-F1	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR

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LE-G5-01	There is no discussion of limitations of severe accident understanding and modeling. This includes such matters as the impact of uncertainty regarding thermally induced SGTR on quantification, the uncertainty of ISLOCA break size and location on timing and source term, and the assignment of CET to end states. Conservative treatment of some phenomena can affect LERF quantification, which in turn impacts LERF and delta LERF results when applying RG 1.174 guidelines in risk-informed changes to the licensing basis, for example.	LE-G5	F	Open	This will be resolved in the next model update to take place before implementation of 4b at PTN. Expected to have little effect on 4b RICTs.	2013 FPR

* 2002 WOG – 2002 WOG NEI 00-02 Global Peer Review

2011 FPR – 2011 Focused Peer Review

2013 FPR – 2013 Focused Peer Review

Table 2 - Fire PRA Peer Review – Findings Resolution

F&O	Discussion	SR	Basis and Recommendation	Disposition in Fire PRA update
1-1	In numerous significant scenarios, the spurious operation probability is assumed to be 1.0 (true) for any events where spurious operation can occur. For example, in scenario 79ALA (one of the top 5 scenarios in unit 3), three events are set to true affecting the top cutsets; GMM0GE100 (MOVs 878A or B spuriously operate), MAVK3CV303A, OHTX3CNTRL. Capability Category I requires setting spurious operation probabilities to industry accepted values. It appears most of the events set to true would be either MOVs (0.33) or AOVs (0.62) or similar, and should not be set to true for significant fire scenarios. Analysis using the specific circuit configuration for each significant spurious operation would be required for CC II, and may lead to different results than the generic values, depending on the circuit design and cable affected.	CF-A1	The overall Fire PRA results appear to be greatly impacted by setting spurious operation probabilities to 1.0. Scenario 79ALA, which is presently 8E-06 would be reduced by at least an order of magnitude by assigning spurious operation probabilities to several events. Similarly, with 79AKA, and 79AJA also at 8E-06. Perform Circuit Failure Probability Analysis for significant spurious operations events, and modify the FRANC model to assign a Perform Circuit Failure Probability Analysis for significant spurious operations events, and modify the FRANC model to assign a probability for the event in the cutsets. In order to meet CCII, the spurious operation probability should be based on the specific circuit configuration for each significant spurious operation.	This F&O has been resolved. At the time of the Peer Review, the FPRA had only a very limited credit for fire induced spurious actuation probability. The specific instance identified in the F&O was updated. The resolution of this F&O also included a review of significant fire initiating events and additional credit for hot short induced spurious operation was applied in the analysis as appropriate. In all cases, the application of the spurious actuation factor is consistent with the guidance in NUREG/CR-6850 and FAQ 08-0047.
1-2	Section 4.1of the Component Selection Report mentions: "Since the FPRA quantification calculates a fire CCDP and the initiating event frequency for each zone is based on the fire ignition frequency, the initiating event faults are not required to be used for FPRA quantification." Fault tree initiating events were not impacted by the component mapping, and are therefore not changed by fire damage. As a result, equipment associated with Fault Tree initiating events were not identified as components potentially causing a fire-induced initiating event. 163 events are screened in Table A of the Equipment Selection Analysis based on being associated a fault tree initiating event. Most are modeled in other system models. However, Several were found to not be modeled in the rest of the model: CPSD3PC611, CPSD4PC611 and 2 related failures. A few others (Cooling units) do not appear to be modeled elsewhere. (This F&O originated from SR ES-A1)	AS-B1 ES-A1 ES-A3 ES-A4 FQ-A2	The significance of not identifying components as causing initiating events is basically that the assumed model impact is accurate by modeling a reactor trip with a subsequent failure of the function, rather than modeling the initiating event itself. In some cases, this impact is a matter of timing for operator actions. In the case of this FPRA, the HEPs have been conservatively set assuming a loss of MFW as a starting point. However, the fault tree initiating events include loss of CCW, loss of HVAC and others. It is not clear that the present model accurately determines CDF/LERF results for systems impacted which may cause a complicated reactor trip (special initiating event). Modify FPRA to model the fire impact to Fault Tree Initiating Events, and analyze the FPRA assuming a fault tree initiating event for those areas where the initiating event can occur.	This F&O has been resolved. The FPRA assumes each postulated fire results in at least a reactor trip. Logic is included in the model so that appropriate event tree is quantified if the fire induces a different type of event (event tree). The overall structure of the FPRA model was reviewed to address the specific item identified in the F&O and to confirm appropriateness of overall treatment. The only change that was required was related to biasing the application of recovery actions so that they were based on an assumed loss of MFW.
1-3	The internal events PRA model has numerous locations in the model where the specific initiating event results in a model impact. For example, under gate U3QT07; initiating events that can cause a PORV or SRV to lift are ANDed with the failure to reclose the PORV or SRV. In this case, special initiator %ZZIP6U3 is identified as an initiating event that will cause a PORV lift, along with %ZZT2U3. Equipment that can cause each are not mapped or modeled in the Fire PRA. As a result of a previous review, the modeling of Feed-and-Bleed was changed to assume a loss of feedwater (low SG level) occurred. The shorter time results in a higher HEP for feed-and-bleed in all scenarios, regardless of whether a loss of FW occurred. However, numerous other modeling impacts can occur, that are not	AS-B1 ES-A1 ES-A3 ES-A4 FQ-A2	As a result of assuming a reactor trip and not mapping components/equipment to modeled internal initiating events; the risk can be under-estimated. In this case, since the general approach used is systematic, this problem is difficult to determine without significant effort to combine the impact of each modeled impact. In most cases, the modeling results in non-conservatism in the result. However, the fix for feed-and-bleed resulted in conservatism for most of the scenarios where FW is not initially lost. In either case; whether modeled conservatively or nonconservatively, the standard requirements in this area are to model the impact of the FPRA accurately. Map all identified internal events initiating events to the specific components that can cause the event, and modify the FPRA to determine the CCDP based on the fire-induced	This F&O has been resolved. The issues and concerns identified in the F&O related to the fire-induced initiating events were reviewed. The review found several instances where a change to the modeling was required to allow the existing treatment methodology to be retained. The review did not identify any instances where specific fire initiating event logic beyond that already in the model was needed.

F&O	Discussion	SR	Basis and Recommendation	Disposition in Fire PRA update
	modeled. Under gate I62115, logic for HVAC unit 3S230 failure to start is included when a Loss of offsite power would occur. This logic is applicable only for when a LOOP occurs, and not applicable for non-LOOP events. This type of logic is contained throughout the internal events PRA modeling. Another example is under gate E1104A, where loss of DC power results in lockout relay failures. There are many other examples throughout the PRA. Additionally, the identification of the specific initiating event for quantification was not performed per the requirements of FQ-A2. For quantification, the modeled initiating event is assumed to be a reactor trip in all cases. This treatment does not meet the intent of SR FQ-A2, where the quantified model should encompass the risk contribution from all applicable initiating events.		initiating event that results.	
1-4	Fire-Induced Spurious PORV opening is modeled under Gate 3S2ORFIREEQUIV (2/6 gate). However, this small LOCA impact is only added to three locations in the PRA model, while small LOCA is located in 11 locations in the CAFTA Model. For example, the fire-induced small-loca logic is not included under gate GHL01 or 7 other locations. For small-small LOCA, the PORV. Similarly, small-small LOCA is included in 9 locations in the PRA model, while the Fire-Induced PORV logic (and other small-small logic) is only included in 3 locations. The above are examples for two of the reviewed added logic. The fire-induced initiator event fault trees have been added into the PRA model. The consistency of the model changes have been reviewed. One question about the changes is that the fire risk could be potentially underestimated. For example, gate 3FIRES2 has been added to simulate the small LOCAs induced by fires. Under gate 3S2ORFIREEQUIV, gate 3FIRES2 and internal event small LOCA initiator %ZZS2U3 are ORed, which seems to be appropriate. However, %ZZS2U3 is under 11 parent gates, while 3FIRES2 only has one parent gate. If the intended fire damage is small LOCA, all system functions affected by small LOCAs should be affected. (This F&O originated from SR ES-B3)	ES-B3 PRM-A4 PRM-B9	Systematic issue of how the PORV, other small-small LOCA initiating events and small LOCA IEs are modeled in the PRA. For any fire-induced initiating events, including PORV opening and other small LOCA initiating events, ensure the newly developed logic is added to all locations in the PRA model to ensure the PRA model solves correctly. Move this F&O to PRM, when evaluated. Also affects the equipment selection, only with regard to the impact of the equipment on the modeled initiating event.	This F&O has been closed. Basic issues involving mapping of the impacts for the fire-induced small-small and small LOCA events were still observed during the focused-scope peer review. These issues are noted in F&O 7-1 which is considered to supersede F&O 1-4.
1-5	HEP EHFCLR3BH is included in the FPRA model, without specific analysis in the task 12 report. This is a modified HEP from the internal events PRA. Additionally, MHFP3BAMT is included in the model as a screening value without documentation in Task 12. (This F&O originated from SR HRA-C1)	HRA-C1	Completeness issue in the documentation of the HRA. Ensure documentation of all HEPs is provided in the HFE report.	This F&O has been closed. The model was changed to remove the events referenced in this F&O. This was reviewed in the focused-scope peer review and was considered sufficient to disposition this F&O.

1-6	Although many of the instruments affecting operator actions are included in Table A-1 of the HRA report, and many are traced, none of the instruments reviewed (partial review) appear to have been tagged as FPRA affecting. As a result, the instruments do not appear to be in the FPRA equipment selection process or equipment list. For example, under GHFPS1RCRC, FI-3-943 is listed and included in the SSA. However, this is listed as non-PRA affecting as is not in the TAGBE, UNL table or other associated table. (This F&O originated from SR ES-C1). Additionally, when new equipment is added that are not in the SSE (for example; LOCAL CHARGING PUMP HYDRAULICA COUPLING TEMPERATURE TI-3/4-6716/6717/6718), these do not appear to be traced or included in the PRA equipment list. Most of the PRA (non-SSE) instruments are not identified by number.	ES-C1 HRA-A3	Systematic issue. As a result of this, the equipment list for the FPRA (considered separate for this review) does not include any of the identified instruments in Table A-1 of the HRA. Include needed Instruments in the PRA equipment list, for later modeling of the impact on the FPRA HEPs.	This F&O has been closed. Table A-1 was modified to indicate credited instrumentation in bold text. This was reviewed in the focused-scope peer review and was considered sufficient to address the issues raised and disposition this F&O.
1-9	Recovery to the Charging Pump Suction valve 114A and the associated components is provided through the excluded events table and the altered events table. However, the recovery is actually crediting flow through MOV-3-350. This flow path is not included in the FPRA model. Similarly, the Normal Containment Coolers are not modeled in the FPRA, but recovered in the altered events table through G174 0.1 recovery. The above are examples. The entirety of the altered events table has not been reviewed for recoveries that may be bringing in additional components not in the FPRA or FPRA Equipment List (This F&O originated from SR CS-A11)	CS-A11 CS-C3 LE-A2 PRM-B15	1st item does not appear to be significant. SSA verifies MOV 350 is available, but the documentation is very difficult to follow. However, the MOV and the associate components do not show up in the Equipment List for the PRA, nor will it show up in the importance lists, results, etc. The surrogate event (0.1 applied to 114A) does not represent what is in the model. Additionally, since the flow path was not modeled, the FPRA considerations may not all be considered. 2nd item: This one may be significant, since the containment cooler can be complicated, and a likely mismatch between the SSA and the modeled FPRA logic could result in significant differences. Add the MOV-3-350 and the associated flow path to the PRA model, normal containment coolers. Additionally, review the altered events report to determine if additional credited flow paths are modeled, where the actual components and support logic is not in the FPRA.	This F&O has been closed. Valve 114A and emergency cooler V30A/B cable routing were incorporated via excluded events based on RFI 0274 routing data. No other exclusion/altered events representing other components were credited in the analysis and no additional issues were noted during this review. The resolution was reviewed in the focused-scope peer review and was considered sufficient to disposition this F&O.
1-10	Transient Fires are postulated in all fire compartments, as listed in Appendix B and Table 3-6 of the Ignition Frequency Report. All factors affecting the fire frequency were assessed based upon a slightly modified NUREG/CR-6850 approach. However, the rankings that were provided do not appear to be consistent with the methods in NUREG/CR-6850, result in an underestimate for fire frequencies in some areas, and an over estimate in other areas. One F&O is provided on this SR. In particular: a) Areas were ranked as zero in maintenance, occupancy, or storage even though entrance to the areas is physically possible, b) Areas were ranked as 1, even though activities were not prohibited by plant procedure. In areas where the room is sealed during operation (roof plugs), transients could have been left in the room prior to sealing, so the ranking on this factor should not be zero - per the 6850 guidance. During the walkdown, Compartments 70 and 71 both had permanently stored	IGN-A9	Systematic issue. Appears as if numerous compartment transient frequencies were underestimated, while others would have been slightly over estimated as a result. Initial review was confirmed by walkdown of 5 areas. The ranking on all 5 areas did not appear to match the walkdown teams estimate for each area. Re-assess the transient fire rankings per the Guidance in NUREG/CR-6850. Confirm the rankings by walkdown of each area, taking into account the actual condition.	This F&O has been resolved. A sensitivity evaluation was performed that involved increasing the weighting factor for occupancy and storage from 'low' to 'medium' for all instances where such a condition could reasonably be expected to occur. The results of this sensitivity found that the impact on the calculated CDF for each unit was less than 1E-7. Given this small impact, the existing analysis is adequate for the application.

	breaker grounding devices, with poly-covers, and 71 had a temporary transformer for the polar crane (operating). Both should be ranked as 'medium' for storage. Similarly, the cable room had storage of 3 temporary fans, cables and blankets and should be marked as medium for storage. This room also appears to include numerous components that will likely be worked on during power, (ranking moderate for non-hot work), and numerous people were present during our limited walkdown. Compartment 88, an open area in front of the switchgear room, had numerous combustibles stored and located, and should probably be marked as medium or high (presently marked as low). Both area 85 and 88 have frequent foot traffic, and should be marked as medium for occupancy. 85 appears as if it should be moderate for storage (no controls). Similarly; no controls appear to be in place for 116. The above are samples of identified issues, based on our limited walkdown. It appears there will be similar issues with other areas in the plant. We looked at other areas adjacent to the areas we were in (compartments 87, 84, etc.), and expect similar problems with the present rankings. (This F&O originated from SR IGN-A9)			
1-11	Events in the altered events table are recovered with a 0.1 failure probability (generally) that include opening of valves, etc. However, these components have not been confirmed as available and are unaffected by spurious operation failing the valve (92-18 concern). (This F&O originated from SR CS-A11)	CS-A11 CS-C3 HRA-D2 PRM-B9	Affects multiple recoveries in numerous compartments Perform 92-18 reviews on all recovered components in the Altered Events Report. Consider also that some recovery events may involve the manual operation of more than one component.	This F&O has been closed. An evaluation supporting the availability of the components recovered via the altered events process was performed, reviewed during the focus-scope peer review, and was considered sufficient to disposition this F&O.
1-17	Table 3-2 includes uncertainty values (EF) for prior and posterior values. However, Error Factors are not propagated to the compartment specific ignition frequencies. The other parameters, such as conditional failure probabilities for circuit failures, do not have uncertainty intervals. The lack of uncertainty intervals would not generate meaningful uncertainty interval of the CDF/LERF results. (This F&O originated from SR IGN-A10)	IGN-A10 QU-E3 UNC-A1 UNC-A2	Systematic Issue. Estimate EFs for significant fire compartments. ESTIMATE the uncertainty interval of the CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.	This F&O has been resolved. The quantitative uncertainty analysis was prepared subsequent to the peer review. A parametric uncertainty evaluation that considers fire ignition frequency as well as other variables was performed that uses a Monte Carlo sampling process. The results of the analysis showed a mean that was slighter higher than the calculated results which was expected.
1-18	During walkdowns, several key areas appeared to have ignition sources not included on the ISDS. For example, in the cable spreading room, 2 transformers were in the compartment (3X033 - 75KVA, 3X130 - 45KVA), both within the screening distance of targets. Also in the compartment is CP-600 spectralink cabinet, an open cabinet, the RCP Vibration Monitoring Cabinet, 4P21 and 4P09 instrument AC panel. Note; we did not do a 100% review of the CS room, so additional cabinets may be missing. See also F&O 1-19. (This F&O originated from SR IGN-A7)	IGN-A7	Appears to be missing components in numerous areas, based on a limited sampling during walkdown. Perform a re-verification of the ISDS for significant fire areas in the FPRA. Add missing components to each ISDS, where applicable.	This F&O has been resolved. The specific instances identified in the F&O were reviewed and the analysis updated accordingly. In addition, the supplemental walkdowns that were performed as part of ongoing analysis refinements efforts for the significant fire areas did not identify any other omissions.

1-19	It appears the Ignition Source Counting did not count Lighting Panels or other similar panels. For example, there were at least 8 lighting panels in the cable spreading room that were not on the ISDS. Additional similar panels are located in most electrical rooms we walked down, such as the switchgear rooms and other electrical rooms. Based on our walkdowns, many of the lighting panels should be included in the ISDS, based on guidance in 6850 and the subsequent FAQ on sealed cabinets. A review of the generic guidance provided for ignition counting did list the screening of small, wall mounted cabinets (sealed). However, the lighting panels do not appear to meet the criteria listed in the procedure (not sealed, numerous switches/breakers), etc. Many of the cabinets are located close to cable trays or other intervening combustibles, so a small fire could result in a larger fire due to spreading. (This F&O originated from SR IGN-A7)	IGN-A7	Appears to be a systematic issue in the FPRA. Include unsealed lighting panels and similar electrical cabinets in the ISDS as potential ignition sources.	This F&O has been resolved. A re-assessment of the lighting panels was performed. The re-assessment focused on the need for treatment as a fire initiating event. No effort was undertaken to alter the population of electrical cabinets considered in the fire frequency development. Therefore, the existing values potentially have a conservative bias. The assessment did not identify any instances were explicit treatment as a fire initiating event was needed.
1-22	Events DACF3ECCB & DACF3ECCA are anded under the containment cooler Fault Tree Logic. Each are applied with a screening value recover of 0.1, resulting in a combined recovery of 0.01 Similarly, AAVK3-2831 and 2832 combine under an AND gate, resulting in a combined recovery of 0.01. The above are only examples, based on a random review of two fire areas, and a few HEPs for each. Two events in about 15 were found to have the above problem. (This F&O originated from SR HRA-C1)	HRA-C1	A review of the Altered Events Report indicates there are likely numerous combinations of scoping HEPs (0.1 in the altered events report. Review the altered events report for significant fire areas, and determine the combined HEPs where they occur under an AND gate. Use a single recovery for these events, or set only one of the events to 0.1 in the altered events report.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is documented in new F&O 7-6 which supersedes this F&O.
1-24	Event EREE3286G3F is listed as 0.0 probability in the altered events report for 2 areas, even though the description says set to .1. A query of the Altered Events Table comes up with 146 entries, where the probability is set to 0.0, but the description says to set the event to 0.1 A question was asked on this issue, and the response was that these were set to 0.0 and another event was set to 0.1. However, there is no way to verify, track or repeat these settings. In reviewing a few events, there was no easy way to determine that there is a corresponding 0.1 event that recovers the failure. (This F&O originated from SR HRA-C1)	HRA-C1 PRM-B11	Multiple discrepancies in the altered events table. Some may be documentation issues (incorrect description). However, many appear to be errors in the probability for these events. Setting the events to 0.0 results in the events being screened from the PRA results. However, many may be as a result of setting other related events to 0.1. In this case, the documentation associated with the reason/description needs to be updated. Correct Altered Events Table, where applicable. Additionally, when events are set to zero, provide a traceable method to determine that a corresponding event is set to 0.1 or another value.	This F&O has been closed. Changes were made to set the altered events to nominal values rather than zero. This was reviewed during the focused-scope peer review and was considered sufficient to disposition this F&O.
1-25	There does not appear to be a review of non-significant cutsets in the PRA documentation. (This F&O originated from SR QU-D5)	FQ-E1 QU-D5	Requirement of QU-D5 as called for by FQ-E1 Perform a review of non-significant cutsets and accident sequences, as discussed in QU-D5 for the FPRA.	This F&O has been resolved. Review of non-significant cutsets performed and documented.
1-26	By using the altered events table, and setting recovered events to 0.1 (scoping) and the other events in the fault tree to zero, random failures that may fail the HEP are not	PRM-A2 SY-A11	Given the large number of events set to zero in the altered events report (2764), and the large number of basic events greater than 1E-03 in the PRA, there are likely numerous	This F&O has been closed. Changes were made to set the altered events to nominal values rather than zero. This was reviewed

	included in the results. If the random events are greater than 2 orders of magnitude below the top or contribute to 1% of the system failure rate, they should be included based on SY-A15. (This F&O originated from SR SY-A15 and PRM-A2)	SY-A15	events not included in the model. Review the values set to 0.0 in the altered events report and modify the solution to include the random failures in the results, when applicable.	during the focused-scope peer review and was considered sufficient to disposition this F&O.
1-27	Significant fire compartment contributors to LERF are documented in Appendix C of the summary report. However, the contribution from plant damage states is not provided or the contributors from LEB SRs. Sources of uncertainty, including sensitivity analysis performed, are not evaluated for LERF. (This F&O originated from SR LE-F1)	FQ-E1 LE-F1 LE-F2 LE-F3 UNC-A1	Requirement of LE-F1, F3. Document the contributors to LERF based on the requirements of LE-F1 of the internal events section of the standard, as required by FE-Q1. Document the Sources of uncertainty, including sensitivity analysis performed for CDF in Appendix D of the Summary Report.	This F&O has been resolved. Added LERF top cutsets and importances run as well as sensitivity analysis in Summary Report. Also performed and documented the uncertainty evaluation for LERF.
1-28	It appears in the analysis that MCC fires for un-vented MCC fires are not considered to damage targets outside the selected MCC. See Scenario 058C-A. The original draft of FAQ 42 included discussion on MCC fires, and provide a probability of the fire propagating outside the MCC. This recommended approach was not included in the final FAQ, and as a result, the treatment of MCC fires does not meet the guidance in the FAQ for a sealed-cabinet. The end result is that the FPRA should consider MCCs as unsealed, due to the possibility of energetic fires resulting in the MCC door being opened, or as a result of maintenance on the MCC being the cause of the fire (actual events) when the door is open at the start of the fire. It is our understanding that the NRC Fire PRA folks also do not consider MCCs to be sealed cabinets. The proposed industry approach basically summarizes to the following: MCCs open at the top are considered unsealed, but MCCs that are sealed at the top have around a 0.1 probability of propagation. The data analysis for this value was not validated, but was considered conservative at the time of the proposal, since several fires where it was not clear if they came outside of the MCC were assumed unknown (1/2 an event). (This F&O originated from SR FSS-A4)	FSS-A4	See FAQ 042. Scenario 058C-A showed a CDF of 1E-05 prior to screening. Include in the model large MCC fires propagating outside of sealed MCCs.	This F&O has been closed. The focused-scope peer review found that the approach utilized is consistent with Final approved FAQ-08-0042, and the F&O was judged to be closed.
1-29	The control room abandonment scenario 106-A, uses a CCDP of 0.1 (scoping value). However, the scenario is one of the top CDF scenarios. (This F&O originated from SR FSS-B2)	FSS-B2	The 0.1 scoping analysis is not based on analysis of plant specific procedures or analysis. The HEP may be higher or lower, depending on procedures. Additionally, the circuit analysis may show the Remote Shutdown Panel may not function for some scenarios due to MSOs. Provide detailed analysis for MCR abandonment CCDP.	This F&O has been closed. MCR evaluations no longer use a 0.1 assumed CCDP. This was reviewed during the focused-scope peer review and was considered sufficient to disposition this F&O.
1-30	Fire modeling was conducted via generic fire modeling from which Zones-Of-Influence (ZOI) for specific initiator types was generated. The ZOIs were used to define bounding fire characteristics for each fire scenario. Characteristics that are used to bound potentially risk contributing fire events are identified in Attachment B of the Fire Scenario Report, (Report 0493060006.004). Based on the use of a bounding	FSS-C1 FSS-G1	The present analysis provides a bounding approach for fire severity in most cases. Perform 2-point fire modeling, when applicable, for significant fire scenarios.	This F&O has been closed. In lieu of 2-point fire modeling, a panel split fraction was used, which is considered an unreviewed analysis method, therefore, the issue is still open. This F&O is superseded by new F&O 10-1 from the focused peer review.

	approach this SR is judged to be met at CC I. Significant fire scenarios should be developed with 2-point fire modeling. (This F&O originated from SR FSS-C1)			
1-31	Fire scenario evaluation tools were developed based on the Generic Fire Modeling Treatments (Project Number SPH02902.030). These walkdown/evaluation tools are based on bounding fires that are assumed to cause target damage at a height above the base fire with the fire burning at peak intensity. Because these tools assume a fire burning at peak intensity this SR is considered met at CC I. (This F&O originated from SR FSS-C2)	FSS-C2 FSS-G1	Significant scenarios are not developed considering fire growth. The current approach is conservative. Include fire growth times for significant fires, where growth time is available in NUREG/CR-6850 or FAQ-052 for transient fires.	This F&O has been closed. In lieu of fire growth timing consideration for fire scenarios, a panel split fraction was used, which is considered an unreviewed analysis method, therefore, the issue is still open. This F&O is superseded by new F&O 10-11 from the focused peer review.
1-32	No evidence was identified that suggests that fires were assumed to burnout over a period of time. Evaluation/scoping was estimated using a peak heat release rate as dictated in Attachment B of the Fire Scenario Report, (Report 0493060006.004). Accordingly this SR is considered not met for CCII/III. (This F&O originated from SR FSS-C3)	FSS-C3 FSS-G1	Significant scenarios do not consider decay. Include growth and decay for significant fire scenarios in the FPRA.	This F&O has been closed. In lieu of fire burnout consideration for fire scenarios, a panel split fraction was used, which is considered an unreviewed analysis method, therefore, the issue is still open. This F&O is superseded and combined with new F&O 10-11 from the focused peer review.
1-33	Except for the MCR fire scenarios, no other fire scenario has used the Non-Suppression Probability (NSP) in PTN fire model at this time. (This F&O originated from SR FSS-D8)	FSS-D8	Systematic. Apply non-suppression factors for significant fire scenarios, when applicable.	This F&O has been closed. This F&O originally identified that no suppression was taken credit for other than for the MCR. PTN now credits suppression outside the MCR, but in doing so, does not evaluate suppression effectiveness properly, as specified in the standard. The issue is therefore, still open and this F&O is superseded by new F&O 10-9 from the focused peer review.
1-34	No evidence was found that supported confirmation of conformance of fire rated barrier segments to applicable test standards. Additionally, the effectiveness, reliability, and availability of any passive fire barrier feature credited does not appear to be performed. (This F&O originated from SR FSS-G4)	FSS-G4	Systematic issue. Provide the documentation that supports confirmation of conformance of fire rated barrier segments to applicable test standards, and the barrier effectiveness, reliability and availability.	This F&O has been resolved. The treatment of barriers in the MCA is based on information in the Fire Hazards Analysis and supplemented with walkdown observations. The analysis documentation was updated to provide this information. The MCA was modified as needed to incorporate the results of this effort.
1-35	The multi-compartment analysis assumes a bounding value of 7.4E-3 for evaluation of active fire barrier elements. Actual fire barrier elements are not considered instead the failure probability of a fire door is assumed for active barrier element failure because this failure probability represents the highest single probability of a barrier failure. Accordingly this analysis provides a qualitative bounding assessment fire barrier feature failure probability. (This F&O originated from SR FSS-G5)	FSS-G5	Systematic issue For any scenario selected if the adjoining physical analysis units are separated by active fire barrier elements, UANTIFY the effectiveness, reliability, and availability of the active fire barrier element.	This F&O has been closed. Issues were identified with the fire barrier assessment for the multi-compartment analysis. This F&O is superseded by new F&O 9-4 from the focused peer review.

1-36	HFEs included in the altered events report are not treated under dependency evaluations Even though the values are screening values, the dependency evaluation may result in a higher HEP, especially if more than 2 events are in a single cutset. The sensitivity study case 3 documented in Appendix D of FPRA Summary Report 0493060006.005 shows that doubling all non-recovery HEPs using multipliers greater than 1 yields a delta CDF increase of 6.60E-5, or 24.7% of the base fire CDF. Furthermore, if doubling the HEPs increased CDF by ~25%, it stands to reason that halving the same set of HEPs would decrease the CDF by a similar amount. It is reasonable to assume that a detailed analysis could reduce most of these screening HEPs by at least half, and in many cases by much more. (This F&O originated from SR FQ-C1)	FQ-C1 HRA-C1 HRA-D2 HR-H3 QU-C1 QU-C2	Systematic Issue Perform dependency analysis for HFEs in the Altered Events Table, and provide a quantification process that incorporates the new HFE dependency.	This F&O has been closed. Dependency between the altered events and between altered events and other HEPs is still not addressed. This F&O is superseded by new F&O 7-8 from the focused peer review which documents the need to address dependency for the events altered to represent recovery actions.
1-37	Significant contributors to Fire PRA results are included in Section 4.3 and the appendices of the Summary Report. This includes a list of operator actions that contribute to CDF. However, no importance measures are provided for CDF or LERF. (This F&O originated from SR QU-D7)	FQ-E1 QU-D7	Requirement of QU-D7 Provide importance measures as required by QU-D7 and FQ-E1.	This F&O has been resolved. Importance measures for CDF and LERF have been determined and added to the Summary Report.
1-38	Results of the Fire PRA did not include the following: (e) the total plant CDF and contributions from the different initiating events and accident classes (i) the uncertainty distribution for the total CDF (j) importance measure results (l) asymmetries in quantitative modeling to provide application users the necessary understanding of the reasons such asymmetries are present in the model (m) the process used to illustrate the computer code(s) used to perform the quantification will yield correct results process. Some of these issues are listed in other F&Os. However, item e(accident classes), l (asymmetries) and m (validation of computer codes) is not covered elsewhere. (This F&O originated from SR QU-F2)	FQ-F1 QU-F2 UNC-A2	Systematic Issue Provide required documentation per QU-F2 and FQ-F1.	This F&O has been resolved. The documentation of the analysis results has been expanded to include the information noted in the F&O. These results were also reviewed for reasonableness and no issues or concerns were identified.
1-40	The quantification of significant basic events, cutsets and accident sequences is not provided. Additionally, the definitions used for significant basic event, significant cutset, and significant accident sequence are not provided. (This F&O originated from SR QU-F6)	FQ-F1 QU-F6 UNC-A2	Requirement of QU-F6 and FQ-F1. Provide the quantification of significant basic events, cutsets and accident sequences, and the definition used for significant basic event, significant cutset, and significant accident sequence.	This F&O has been resolved. The Summary Report has been updated to provide the importance measures of the model basic events, top 90% of all plant cutsets, and a review of the scenarios contributing more than 1% of the total risk.
1-41	The HRA does not look at the Fire Specific factors affecting: (a) quality [type (classroom or simulator) and frequency] of the operator training or experience (b) quality of the written procedures and administrative controls (c) availability of instrumentation needed to take corrective actions (d) degree of clarity of cues/indications (e) human-machine interface (f) time available and time required to complete the response (g) complexity of the required response (h)	HRA-D2 HR-H2 PRM-B11	The simplified factors included in the HEP modifier approach does not provide the equivalent of a detailed HRA. Provide a detailed HRA for significant HEPs in the FPRA results. Detailed HRA should account for the Fire-Specific factors as listed above. The present multiplier method does not appear to meet the requirements of the standard for detailed HRA.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of

	environment (e.g., lighting, heat, radiation) under which the operator is working (i) accessibility of the equipment requiring manipulation (j) necessity, adequacy, and availability of special tools, parts, clothing, etc. As a result, the HRA does not appear to meet the PRA standard requirements for a detailed HRA. (This F&O originated from SR HR-H2)			documentation for screening values retained and detailed HEP development for significant HFEs is documented in new F&O 7-6 from the focused peer review, which supersedes this F&O.
1-44	The FPRA models 0.1 HEP values for several recovery actions that are not in the SSA and are not in the fire safe shutdown procedures (See F&O 6-11). These recovery actions are input into the PRA model by adjusting the random independent failure probability for the recovered component to 0.1 and the logical true setting for components that prompted the recovery to 0.0. This approach introduces several issues, including the following: The use of 0.0 values that are intended to account for recovered equipment eliminates the 0.1 recovery HEPs altogether in several instances. For compartment 070-AB, basic event EC8R330303 is set to 0.1 for recovery of alternate feed to load center 3H, but this recovery feeds into AND gate E3013H. This AND gate appears to be nullified, however as the second input to the gate is false due to the 0.0 values present in altered events (based on visualization of settings from the altered events file for 070-AB in CAFTA). A quantification of 070- AB produced 10,000 cutsets and no instance of EC8R330303 was present. A further review of the 070-AB quantification indicated the following events placed in the altered events table as recoveries do not appear in the final cutsets: ATPXPUMPASTRT, EB2F33003H, ECBR33AA15, FAVC3-1606, MAVX3-311, MSVR3-311. Further review of other areas and other recovery values (0.1 and 0.0 values in the altered events report) appears to indicate similar problems will occur. The above are examples (potential issues), which appear to be logic problems resulting from the use of 0.0 and 0.1 inputs in the altered events report. Additional problems are likely for other events. (This F&O originated from SR HRA-C1)	HRA-C1 PRM-B11	Appears to be a significant non-conservative impact to PRA results. Given the actions in the altered events report are being added to the model as needed recoveries in order to ensure risk is low, and given the resulting recovery actions do not show up in the results in most cases, there appears to be a disconnect between the addition of new actions to the procedures and the quantification of these actions in the FPRA. It appears part of the disconnect is that the logic modeling, as modified by the altered events table, results in the recovery values being screened from the results. Revise the approach described for assigning recovery HEPs via the altered events table and ensure the quantification produces the intended results.	This F&O has been closed. changes were made to set the altered events to nominal values rather than zero. This was reviewed and is considered generally sufficient to disposition this F&O. However, one additional issue was discovered during this review which is documented in F&O 8-10 from the focused peer review, which supersedes this F&O from the 2010 peer review.
1-45	The method and calculations for transient fire severity factors (SF) are not clearly documented and several SF values in FRANC model are not consistent with the ones listed in the FSS report. For example, fire scenarios 079A-J/K/L in Appendix A of FSS report have an SF value of 3.05E-2, which is used in the FRANC model, but is not consistent with the 8E-2 value included in FSS report section 8.4, GENERAL TRANSIENT SEVERITY FACTORS. FPL/ERIN staff reviewed this issue and stated that the SF is calculated based on a floor area factor (FSS report section 8.3) times the 8E-2 transient ignition frequency adjustment factor (FSS report section 8.4). For	FSS-D3	The Severity factors used are in significant fire scenarios in the FPRA. Document the severity factors used for each scenario including the basis. Revise the transient severity factors to remove double counting of the area factor included in both the square footage of the compartment, and the 8E-02 calculated in Section 8.4 of the FSS report.	This F&O has been closed. Transients now only use floor area ratio. Transients no longer apply SF and NSP factors which may have resulted in non-conservative results. This was reviewed during the focused peer review and is considered sufficient to disposition this F&O.

	079A-J/K/L, this method results in a factor of 1.52E-2. A factor of 3.05E-2 was used in FRANC. The above points to the following: a) The severity factors used are not well documented, traceable, or consistent with what is provided in the text of the FSS report. B) The two severity factors basically double count the area severity factor. For the last case (B), the 8E-02 already includes a consideration, based on experience, that the fire that occurred is near a target (component). As a result, the 6 events listed just happened to not be near any components, resulting in a low probability of damage. To put this another way, it can not be demonstrated that the 8E-02 factor is due to the small size of the transient fires rather than being due to the fire being in a location not near a component. A review of the control/aux bld transient fires was performed. The first event caused an automatic suppression system actuation, which indicates a fairly large fire occurred. It takes a pretty good size fire to raise the fire detectors up to 160+ degrees. A second event is described as "A leaking regulator ignited leaking propane." This can obviously be a larger fire. Since the location/area of the originating fire is in the 8E-02 factor, and in the "area" factor, the double counting results in an underestimation of the likelihood of fire damage for a transient fire. (This F&O originated from SR FSS-D3)			
1-46	The PTN FPRA methodology generally does not include postulation or evaluation of smoke damage. Additional review shows that the smoke issues do not affect the FPRA results significantly. However, the FPRA does not include a qualitative evaluation of smoke damage to FPRA equipment. (This F&O originated from SR FSS-D9)	FSS-D9	Requirement of FSS-D9 Perform a qualitative evaluation of smoke damage to FPRA equipment per the requirements of FSS-D9.	This F&O has been closed. Smoke damage generic methodology has been added to section 6.2 of the fire scenario report, however, no results of the assessment are documented, and no evidence of smoke failures in the scenarios. This issue is superseded by new F&O 10-12 from the focused peer review.
2-1	The plant partitioning task does not include detailed discussion with respect to this SR PP-B7. The manholes are modeled as separate fire compartments. However, no walkdown for these manholes has been performed. No justification for the modeling approach has been provided except being briefly mentioned in Section 2.2 of Report PTN-PSA-7.01 Revision 2. Walkdowns were also not documented for spatial separation or other boundaries that are not fire rated but was credited in the FPRA. (This F&O originated from SR PP-B7)	PP-B1 PP-B7	Section 3.11.5 of FHA states that man-hole covers are justified as three-hour fire boundary although they need not to be specifically rated as fire barrier. Therefore, the modeling of manhole as fire compartments is considered acceptable although no walkdown has been performed for the manholes. Other credited barriers are discussed in PP-B2-4 above Consider adding justification for the modeling of manholes according to the requirements in SR PP-B7. Consider performing walkdown for manholes with significant risk contribution. Also, document walkdowns on all credited, nonrated barriers credited in the FPRA.	This F&O has been resolved. Walkdowns of fire zone boundaries were performed and documented in support of a review of the Fire Hazards Analysis update. Additional discussion regarding the basis for the ignition frequency for the manholes was added to the documentation.
2-4	The FRANC model itself cannot generate total CDF / LERF and risk importance measures. The models show the contributions from each quantified fire sequences. As shown in the summary report 0493060006 Rev 1, the Unit 3 CDF top cutsets are listed. However, the Unit 4 CDF, Unit 4 CDF & LERF, the risk importance values for each basic	PRM-A3	This finding mainly focuses on documentation. However, the unavailability of the cutset files and risk importance reports prevents the detailed analysis in other tasks such as HRA and circuit analysis, etc. Document cutset files for U3 LERF and U4 CDF/LERF and risk importance reports for each.	This F&O has been closed. Aggregate CDF and LERF cutsets were provided during the focused scope review that address the concerns in this F&O. Results and importance measures for Unit 4 were also provided in the TURKEY POINT NUCLEAR PLANT FPRA

	events are not included. (This F&O originated from SR PRM-A3)			SUMMARY REPORT NUREG/CR-6850 TASK 16 (Report 0493060006.005 Revision 2). F&O 8-8 from the focused peer review supersedes F&O 2-4 and documents the need to ensure the comparable Unit 3 results are documented.
2-6	Conservative screening values were used for risk-significant human actions, including both the new fire specific safe shutdown actions identified and the non-fire actions that exist in the internal events PRA and included in the Fire PRA. It is noted that the HEPs derived by the multiplier approach for the non-fire actions are still considered as the screening values until a detailed analysis is performed. There are numerous examples of significant HEPs in the results, as well as significant screening HEPs set to 0.1 in the altered events report. The estimation of the HEPs for the new, fire-related human actions did not account for the fire scenario-specific performance shaping factors (PSFs), including time available to complete action, etc. The formal HRA of significant HEPs includes referencing and considering procedures to perform actions, assessment of the impacts instrumentation needed for cues and execution and resulting impacts to the HPE, feasibility, timing of the event, performance shaping factors, evaluation of cognitive and execution error probabilities, scenario-specific equipment impacts that may affect the timing of the human interaction, as well as considerations of workload (for input to the dependency impact evaluation). Overall, it appears as if the Fire PRA treats the screening results of the method used for the non-fire human actions as detailed results. However, as indicated in the diagram shown in the report, the method is a "simplified" method, and does not meet the requirements of a detailed HRA per the standard. Although the original HEPs in the internal events PRA included all of the relevant HRA factors, the fire-specific HEPs would have to consider the impact of fire on these factors in developing the HEP results. The end result is an HRA that provides very general results, without specific application to a fire compartment or scenario, and a lack of detailed HRA for significant fire areas. (This F&O originated from SR HRA-C1)	FQ-C1 HRA-C1 HRA-D2 HRA-E1 HR-G1 HR-G2 HR-G3 HR-I1 HR-I2 PRM-B11 QU-C1	To meet CC-II requirements, the risk significant HFEs should be evaluated in more detail, as specified in HRA-C1. The approach used to estimate HEPs for the risk-significant, new fire-related operator actions should use an approach that addresses both failure in cognition as well as failure to execute. Update the HRA by performed detailed HRA for all risk-significant HEPs, including the HEPs analyzed using the multiplier method in the HRA, as well as the screening HEPs in the altered event report. The HRA approach should be revised to treat the simplified approach for the non-fire actions as a screening tool, and provide detailed HRA for significant HEPs. Additionally, even the screening results should be reviewed for each fire scenario where the HEP is applied, including consideration of timing, lost indications, spurious operations in the scenario and other effects on the timing for the HEP.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is documented in the focused-scope peer review F&O 7-6 which supersedes this F&O from the 2010 peer review.
2-7	PTN system model changes in the fire PRA models are summarized in Tables D-1, D-2 and D-3 of the Component and able Selection Report 0493060006.001, Revision 1, Appendix D, MULTIPLE SPURIOUS OPERATIONS EXPERT PANEL EVIEW AND DISPOSITION OF OPEN ITEMS. However, no additional documentation of the changes for PTN fire PRA has been provided. Requirements under SY-A and SY-B are not met as a result. In particular, the requirements of SY-A2, 3, 4, 6, 11, 14, 15, 17, and 23, as well as SY-B5 through B15, do not	PRM-B9 SY-A11 SY-A14 SY-A15 SY-A17 SY-A2 SY-A23 SY-A3 SY-A4 SY-A6	Systematic issue. System models were not updated according to the SY-A and SY-B SR requirements. Consider updating the system models and their associated documentation according to SY-A and SY-B SR requirements.	This F&O has been closed. Changes made to the model to incorporate fire impacts are addressed in the PTN PSA Model Update Calculation, PTN-BFJR-00-001, Revision 9. The documentation provided is consistent with the process normally used for PSA model updates and is considered sufficient to disposition this F&O per the focused-scope peer review.

	appear to be met based on a review of the documentation in the tables. Since the level of analysis and documentation for system models for Fire PRA is expected to be similar to that performed for internal events, the documentation in the Appendix D tables does not meet what is expected for this requirement. For example: on Table D-3, page D-57, there appears to be a number of changes to 'correct' logic. The original logic is included in the system notebooks (e.g., DC power notebook). However, since this is just corrected here, the internal events PRA is not updated. Additionally, without proper development, it is impossible to determine if the change is accurate. In the DC power case, a reference to the power drawing is needed to ensure the logic is now correct. On Table D-2, item 23, there are a lot of logic changes. However, there is no reason as to why the logic changes are needed or why the changes are made as written. In Tables D-2 and D-3, numerous references have been made to the comments / recommendations from site engineers, which add insights but should not be considered as the sole modeling basis. (This F&O originated from SR PRM-B9)	SY-B10 SY-B11 SY-B12 SY-B13 SY-B14 SY-B15 SY-B5 SY-B6 SY-B7 SY-B8 SY-B9		
2-10	A review of the HRA report and the recovery rule file used in the FRANC model shows that the majority of the HEPs and HFE combinations were treated properly in the PTN fire PRA model. However, isolated cases indicate the following issues: 1. Some HEPs were not applied (or documented) properly. For example, EHFPDOSTXT, Failure to cross-connect unit diesel oil storage tanks to extend availability of fuel for EDG, was supposed to be set to 1.0 per Table A-2 in HRA report page A-19. However, it is set to 2.3E-2 in the recovery rule (in both the HRA report and actual rule file). A review by FPL/ERIN staff showed that the example HEP was included in the UNL table (failed for all scenarios), however, it was also included in the Excluded Events table for multiple fire scenarios. Since this HFE has no cue in the main control room, it was intended to fail this HEP for all fire scenarios. Therefore, the Excluded Events table should be updated. 2. A review of the top cutsets in fire sequence 096A show the HFE combination CHFPSTPRCP and GHFPINJVLVS has not been considered in the HRA evaluation. FPL/ERIN staff concurred that some HEP combinations may be missed, which render conservative results. Because fire scenarios for fire compartment 096 will be refined for realism, the HEP combinations are expected to be re-visited. (This F&O originated from SR PRM-B11)	PRM-B11	One of the identified issues generates non-conservative results and the other one is conservative. The second example is likely significant. Consider reviewing the recovery rule file for consistency against the HRA report documentation. Also consider updating the HRA combination evaluation.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-12	A comparison of basic events between the internal event (Rev. 7) and fire models has been performed. For the new basic events (about 650 basic events are identified) in fire model, the majority is set to either 0 or 1, which simulates the fire impact in the fire scenarios. The ones with other	PRM-B12 PRM-B13	Isolated error. However, the error may be significant Consider providing basis for the modeled new basic events in the fire PRA model. Correct AHFPAFWFLO probability in FRANC.	This F&O has been closed. he action taken to address this item was specifically included in the focused-scope Peer Review.

	values are checked and found to be lacking sufficient basis as required by SR PRM-B13. For the modified HEPs, the majority seems to be ok by updating with a more bounding value of 1.0. However, the following event is an outlier, which may result in early truncation of the cutsets with this AHFPAFWFLO, OPERATOR FAILS TO THROTTLE UP AFW FLOW, 1.7E-4 For the deleted basic events in fire model, all the changes have been traced in the PRM report. (This F&O originated from SR PRM-B13)			
2-14	Fire compartment 096 is the top contributor for U3 CDF. Tracing the failed basic events / components /cables shows that the sequencer failures seem to contribute to loss of redundancies. Since the cables travel to rooms housing redundant trains, the assumed failure of cables fail the sequencer, which in turn fails the switchgears. For example, circuit analysis for 3X03-NPO-3AA states, "ADDED CABLES IN SEQUENCER THAT CAN PREVENT SUT FROM POWERING SWGR THRU 3AA05 BREAKER." A discussion with FPL/ERIN staff indicated that the analysis for fire compartment 96 is not yet complete as documented in the summary report. It is anticipated that the approaches and refinements used for other plant locations will result in more realistic risk results for the room. On the other hand, FPL staff also identified that there is an open item associated with fire zone 096 (SSA-3GG-13). (This F&O originated from SR PRM-B9)	PRM-B9	Current analysis for top fire scenario 096A is conservative. This issue applies to other fire scenarios in the FPRA. Consider updating the top fire scenarios to remove conservatism related to sequencer modeling and failures.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-15	Most new events are added to the model in order to assess spurious operation, and other Fire PRA effects. However, there is not documentation supporting the events, and as such, they do not meet the DA requirements as referenced in PRM-B13. This SR lists an exception (DEVELOP a defined basis to support the claim of non-applicability of any of these requirements in Section 2), which is not provided. (This F&O originated from SR PRM-B13)	PRM-B13	Systematic Issue. Consider adding to the documentation whether any events added to the fire PRA in Table D-1, 2, or 3 of the component selection report are new to the PRA. If new, add the details of the event to the documentation per the associated standard requirements or provide justification to the PRA documentation of non-applicability of any DA SR.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-16	RCP Spurious start logic under 3FIRERCPSPUR3 appears to be incorrect. First, the spurious start of an RCP typically takes 2 spurious operations, including start of the lift pump, and then start of the main pump. Second, the logic does not include any consideration of whether the operator already tripped the pumps, and they restarted or whether the RCP received a spurious signal, and then operator trip of the pump is not possible (may be possible with a single spurious depending on the design). Finally, if the RCP restarts (as modeled), tripping of the RCPs may not be possible depending on the operator actions performed. This consideration needs to be included in the determination of the event for operator trips the RCPs. Under gate 3FIRERCPSPUR1, 3FIRERCPSPUR1 (RCP Seal Failure	PRM-B9	Isolated issue. Risk impact is not known Consider refine the model for the spurious start of RCPs. Also address the fire impact of the operator action to trip the RCPs.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.

	Following Spurious Pump Start Due to Fire) is AND'ed with the HEP CHFPSTPRCP, which is not failed by any fire scenario. This event is evaluated in the HRA report, but the fire impact for an operator to trip the RCPs is not evaluated, (This F&O originated from SR PRM-B9)			
2-18	PTN LERF model development is documented in the Component and Cable Selection Report 0493060006.001, Rev. 1, Section 4.5, LARGE EARLY RELEASE FREQUENCY (LERF). This section described the excluded LERF sequences in internal events model. No discussion is identified on any new accident progression beyond the onset of core damage that would be applicable to the Fire PRA that were not addressed for LERF estimation in the Internal Events PRA. As a result, significant accident progression sequences resulting in a large early release have not been reviewed to determine if repair of equipment can be credited for the FPRA LERF models [LE-C3 CC-II requirement]. In addition, accident sequence dependencies in the accident progression sequences have not included in a manner consistent with the applicable requirements of 2-2.2, as appropriate for the level of detail of the analysis [LE-C8 requirement]. (This F&O originated from SR PRM-B14)	LE-C1 LE-C3 LE-C8 PRM-B1 PRM-B14	Systematic issue. Identify and document any new accident progression beyond the onset of core damage that would be applicable to the Fire PRA that were not addressed for LERF estimation in the Internal Events PRA. REVIEW significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. JUSTIFY credit given for repair (i.e., ensure that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability [see SY-A24, DA-C15, and DA-D8]). INCLUDE accident sequence dependencies in the accident progression sequences in a manner consistent with the applicable requirements of 2-2.2, as appropriate for the level of detail of the analysis [LE-C8 requirement].	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-19	The PTN fire LERF model is built upon a draft internal events LERF model (rev. 8), which needs to be updated when that model is finalized. (This F&O originated from SR PRM-B14)	LE-A1 LE-A2 PRM-B14 PRM-B15	Draft internal event LERF model is used. Changes to the draft model when finalized could be significant. Update the fire LERF model when the internal event LERF model is finalized.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-22	Beyond the Generic Fire Modeling Treatments, the Fire PRA did not include additional detailed fire modeling for most fire compartments. (This F&O originated from SR FSS-C6)	FSS-C6 FSS-D3 FSS-G1	Without detailed fire modeling for significant fire scenarios, the results are conservative. Consider performing additional detailed fire modeling for target damage timing when the exposure environment exceeds the damage threshold.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-26	The system unavailability records for the plant have not been reviewed in crediting fire detection and suppression systems. (This F&O originated from SR FSS-D7)	FSS-D7 FSS-D8	Systematic issue. The intent for Capability Category II is to additionally require a review of plant records to determine if the generic unavailability credit is consistent with actual system unavailability. Outlier experience would be any experience indicating that actual system is unavailable more frequently than would be indicated by the generic values. Consider performing and document the review of plant records to determine if the generic unavailability credit is consistent with actual system unavailability. Outlier experience would be any experience indicating that actual system is unavailable more frequently than would be indicated by the generic values.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-29	PTN FSS report 0493060006.004 Revision 1, App. A documents the SCENARIO SUMMARY REPORT, which includes the combinations of fire sources and target sets. However, the walkdown documentation is lacking.	FSS-D10 FSS-D11 FSS-H10	Documentation issue. However, the inadequacy of the walkdown documentation cannot provide detailed information for scenario development or detailed fire modeling. Consider enhancing the process and documentation of the source-	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer

	Currently, the walkdown results are documented directly in the FRANC database. Consider a walkdown documentation package which would include elements such as a data collection procedure, documentation of who performed what walkdowns on what dates, documentation that review of the collected source-target data was performed, etc. (This F&O originated from SR FSS-D10)		target data collection walkdowns. See recommended walkdown information in NUREG/CR-6850.	Review.
2-31	The PTN fire PRA model has not completed the quantitative results for any scenarios analyzed quantitatively in a manner that facilitates Fire PRA applications, upgrades, and peer review. (This F&O originated from SR FSS-H8)	FSS-H8	Quantitative results for the identified significant fire scenarios in the multi-compartment analysis should be generated and documented. Quantify the identified significant fire scenarios in the multi-compartment analysis and complete documentation.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-37	The fire PRA has not identified how the physical characteristics (unique to fire scenarios, or affected by fire scenarios) identified in LE-A1 and the accident sequence characteristics identified in LE-A2 are addressed in the LERF analysis. FPRA impact can affect accident sequences for LERF such as failing containment isolation, affecting containment cooling or instrument air to containment, opening pressurizer PORVs (pressurizing containment), etc. (This F&O originated from SR LE-A3)	LE-A3 PRM-B15	Requirement not met. Identify how the physical characteristics (unique to fire scenarios, or affected by fire scenarios) identified in LE-A1 and the accident sequence characteristics identified in LE-A2 are addressed in the LERF analysis.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-38	The fire PRA did not re-visited the plant damage states defined in the internal events LERF model. (This F&O originated from SR LE-A5)	LE-A5 PRM-B15	Requirement not met. Re-define the plant damage states in the internal events LERF model to account for any fire-specific characteristics	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-39	The MSO Review List reviewed by the Turkey Point Expert Panel should have captured most of the requirement in this SR. However, a systematic process and documentation are not available with respect to the identification of LERF contributors from the set identified in AMSE standard Table 2-2.8-9. (This F&O originated from SR LE-B1)	LE-B1 PRM-B15	Systematic issue. Use a systematic process and document the identification of the fire-specific LERF contributors from the set identified in AMSE standard Table 2-2.8-9.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-40	A separate FPRA LERF package is not available. SR LE-C2 is assigned as not met since the LERF specific operator actions is not evident in the HRA report and the screen values are used for numerous operator recovery actions for the fire-induced component failures. (This F&O originated from SR LE-C2)	LE-C2 PRM-B15	LE-C2 CC-II requirement INCLUDE REALISTIC treatment of feasible operator actions for LERF-Specific HEPs (if any) following the onset of core damage CONSISTENT WITH APPLICABLE PROCEDURES.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-42	FPRA specific significant accident progression sequences resulting in a large early release have not been developed. The FPRA LERF changes are directly incorporated in the quantification fault tree. FPRA specific realistic generic or plant-specific analyses for system success criteria for the significant accident progression sequences have not been developed. The FPRA LERF changes are directly incorporated in the quantification fault tree. FPRA specific	LE-C4 LE-C5 LE-C6 PRM-B15	SR LE-C4/5/6 Requirements For FPRA LERF model, evaluate the fire-specific aspects for the following requirements in SRs LE-C4 through C6:INCLUDE model logic necessary to provide a realistic estimation of the significant accident progression sequences resulting in a large early release. INCLUDE mitigating actions by operating staff, effect of fission product scrubbing on radionuclide release, and expected beneficial failures in significant accident progression sequences.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.

	system models that support the accident progression analysis have not been developed. The FPRA LERF changes are directly incorporated in the quantification fault tree. (This F&O originated from SR LE-C6)		PROVIDE technical justification (by plant-specific or applicable generic calculations demonstrating the feasibility of the actions, scrubbing mechanisms, or beneficial failures) supporting the inclusion of any of these features. USE appropriate realistic generic or plant-specific analyses for system success criteria for the significant accident progression sequences. USE conservative or a combination of conservative and realistic system success criteria for non-risk significant accident progression sequences. DEVELOP system models that support the accident progression analysis in a manner consistent with the applicable requirements for 2-2.4, as appropriate for the level of detail of the analysis.	
2-43	The significant accident progression sequences resulting in a large early release have not been reviewed to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. The significant accident progression sequences resulting in a large early release have not been reviewed to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. (This F&O originated from SR LE-C10)	LE-C10 LE-C12 PRM-B15	LE-C10/12 CC-II requirement REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences. REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	This F&O has been closed. The action taken to address this item was specifically included in the focused-scope Peer Review.
2-44	Uncertainty Evaluations (Sensitivity studies) should be performed for both CDF and LERF model for Units 3 and 4 since the model uncertainties may have different impact to specific model due to differences in plant designs, FPRA model details, and etc. (This F&O originated from SR QU-E4)	QU-E4 UNC-A1 UNC-A2	QU-E4 requirements. Perform sensitivity studies should be performed for both CDF and LERF model for Units 3 and 4.	This F&O has been resolved. Parametric uncertainty and sensitivity has been performed for CDF and LERF for both Units. The results do not indicate any change in the selection of parameters or assumptions are necessary.
3-2	Credit for fire compartment separation via non-rated construction was commonly noted, e.g., according to the FHA the walls of fire compartment 034 are not fire rated and they provide separation from fire compartments 036, 035, & 058. Separation of FC 034 from the surrounding FCs is one of many examples where non-fire rated construction is credited for separation. Use of this level of separation is acceptable provided the separation is justified. However, the justification does not appear to be provided for the FPRA. (This F&O originated from SR PP-B2)	PP-B1 PP-B2		This F&O has been resolved. The configuration and construction of non-fire rated barriers was confirmed using a combination of information in the Fire Hazards Analysis and supplemental plant walkdowns. The analysis and related documentation was updated to provide this information.
3-3	A few cases of special separation are credited in the PB&P. Most notable are separation of Fire Compartments 058 and 037 and 004 and 010. The FHA notes in the write-up for fire zone 004: 'There is a partial height concrete wall on the South side of this room with a full height opening to Fire Zone 10'. No justification is provided for this separation,	PP-B1 PP-B3	Two instances were identified where spatial separation is credited for the separation of fire compartments. No justification is provided for this separation. Provide justification for the use of spatial separation in the FPRA. If not justified, combine the compartments in the FPRA.	This F&O has been resolved. Openings between fire zones were addressed with respect to targets on the other side of an opening which are within the zone of influence of an ignition source. Targets were evaluated for fire damage regardless of the zone in which they were located.

	hence it is not clear that the credited separation may be expected to contain the effects of a fire. Accordingly the effect of a fire beyond the identified fire compartment boundary may occur. While this effect would be expected to be identified through performance of the multicompartment analysis the level of documentation provided in support of the PB&P does not satisfy the standard requirements. (This F&O originated from SR PP-B3)			The multi-compartment analysis considered the volume associated with adjacent zones with openings between the zones in evaluating the potential for hot gas layer formation.
3-4	The PTN self assessment points out that the FHA documents the use of active fire barrier features as necessary for fire zone separation. However in cases where fire compartment separation is provided by unrated barriers there may be active features that are not identified by the FHA but credited by the Fire PRA. In such cases active fire barrier features may be unknowingly credited for separation but not adequately maintained by the fire protection program. Because these elements were not purposely identified within the development of the Fire PRA it is unknown if the Fire Protection Program identifies all of the necessary features. Because the Fire PRA does not formally define and justify these features this element is judged not met. (This F&O originated from SR PP-B5)	PP-B1 PP-B5	As discussed in the description justification/discussion is not provided for crediting active fire protection features in barriers that are identified as non-fire rated structures. It is not clear if active features such as fire dampers exist in these barrier segments because the FHA does not rely on them for separation. Documentation should be provided that clearly establishes what features are credited in such barrier segments and why makes them acceptable. Given the large number of barriers credited in the FPRA that are discussed in the FHA, but without discussion of active elements, there are likely a number of undocumented active elements in these barriers. Determine the active fire barriers on barriers credited in the FHA (not SSA), and provide justification for any active elements credited in the FPRA.	This F&O has been resolved. The walkdowns that were performed did not observe any open fire doors (active features). The documentation for the fire scenario development process was updated to provide the criteria and methodology that were used.
3-5	According to the Section 3.13 of the PTN FPRA Summary Report the effect of an earthquake on ignition source scenarios is discussed in the IPEEE and Potential Fire Related Vulnerabilities self assessment. Review of the Potential Fire Related Vulnerabilities self assessment did not reveal an analysis that specifically addresses generation of fire ignition source scenarios which could result from an earthquake, nor does this assessment address the potential risk significance of these scenarios. This assessment does identify fire vulnerabilities in terms of fuels, ignition sources, and oxidizers however these discussions are not specific to seismic events nor do they include evaluation of special ignition scenarios that may arise from an earthquake. (This F&O originated from SR SF-A1)	SF-A1	As discussed in the description no discussion was found that specifically addresses fire ignition source scenarios that may arise from an earthquake. Also, since these scenarios are not identified a qualitative assessment of their risk significance is not included. The analysis provided in the Potential Fire Related Vulnerabilities self assessment should be expanded to look for unique ignition source scenarios that may arise from an earthquake and a discussion of the risk significance of these scenarios should be qualitatively assessed.	This F&O has been resolved. The low seismic spectra applicable to the Turkey Point site have been validated via the IPEEE with respect to the potential for causing unique fire scenarios. Their potential for causing damage to pipes or tanks containing combustible gases or liquids or to initiation of electrical fires is considered negligible.
3-7	According to report PTN-PSA-7.01 The generic fire ignition frequencies provided in NUREG/CR-6850 were used to establish the fire ignition frequencies for PTN. While the use of these values is not entirely incorrect, this SR requires the use of 'current nuclear power industry event history that includes power plants of similar type, characteristics, and vintage.' Accordingly this requirement requires use of the EPRI revised generic fire frequency values included in FAQ 08-048 or justification for its exclusion. Also, it appears that FAQs 07-35 (bus ducts) and 08-44 (MFW pump fires) were not incorporated into the FPRA. (This F&O originated from	IGN-A1 IGN-B4	As discussed in the description the revised generic fire frequencies contained in FAQ 08-048 are not incorporated into the PTN fire frequencies nor is there justification for their exclusion. This SR requires use of the current nuclear power industry event history or justification for data exclusion. Because the fire ignition frequency methodology does not address the data contained in FAQ 08-048 this SR is considered not met. Use of the NUREG/CR-6850 values results in a conservative estimate of CDF/LERF. FAQ 35 can have significant impact on fires in the area of bus ducts. However, it is not apparent if this is important for Turkey Point. FAQ 44 can result in a lower MFW large fire frequency. The	This F&O has been resolved. The guidance provided in FAQ 08-0048 requires the use of the original NUREG/CR_6850 fire frequency values as a sensitivity study. Rather than perform two analyses, the PTN analysis was developed using those original values for the NFPA 805 application. The application of the non-segregated bus duct information from FAQ 07-0035 is not applicable as the plant does not use non-segregated bus duct. The connections to the station transformers are made using cables. FAQ 08-0044 was also not

	SR IGN-A1)		fire ignition frequency information contained in FAQ 08-048 should be incorporated into the PTN fire ignition frequencies. Additional FAQs should also be incorporated into the FPRA.	needed and the conservatism associated with original method did not adversely affect the results.
3-8	Review of the plant-specific fire events for outlier experience indicates that some events may have been considered outliers or unknown if the selection criteria had considered treatment of fires that are extinguished prior to full development as potentially challenging. Several cases identified in Appendix A of the Fire Ignition Frequency Development Report, PTN-PSA-7.01 may have developed into challenging fires had they not been discovered and extinguished early in their development. Fires 7, 8, 9, 21, 22, 27, 30, 31 appear to be potentially challenging fires (or unknown). See also the previous assessment from 9-09. (This F&O originated from SR IGN-A4	IGN-A4	As discussed in the description review of the fires identified in Appendix A reveals fires that may have become challenging had they not been extinguished early. The selection criteria for challenging fires contained in Appendix A is based on section C.3.3.1 of NUREG/CR 6850, however the criteria contained in C.3.3.2 is not included; had the criteria of C.3.3.2 been included more fires may have been selected as challenging or identified as unknown. The criteria for selecting challenging fires in Appendix A of the Fire Ignition Frequency Development Report, PTN-PSA-7.01 should be revised to include the criteria contained in C.3.3.2 of CR/NUREG 6850 and the fire events should be revisited to determine if additional fires should be selected.	This F&O has been resolved. The scope of plant specific fire events were re-assessed with an expanded group of plant personnel with particular focus on the subjective criteria from C.3.3.2. The results of the re-assessment affirmed the previous dispositions.
3-9	One situation was identified for which credit of a fire wrap is proposed. The FRANC Excluded Events Table, Attachment E of the Fire Scenario Report, (Report 0493060006.004) indicates that a fire wrap will be credited in Fire Zone 071. This fire wrap protects an MCC 3B cable and is being credited to exclude basic event 3B06. No technical basis for the fire resistance rating of this wrap was found in the FPRA nor is there justification for crediting this wrap assuming mechanical damage, direct flame impingement, or HEAF. Accordingly this SR is considered not met. During the walkdown, Thermo-lag was seen throughout the plant. Thermo-lag has had problems in the past, and the rating would need to be justified prior to credit. (This F&O originated from SR FSS-C8)	FSS-C8 FSS-G1	This finding is based on identification of credit for a wrap in the FRANC Excluded Events Table, Attachment E of the Fire Scenario Report, (Report 0493060006.004). Credit for the proposed fire wrap should be addressed in the wrap integrity should be established with respect to fire resistance, mechanical protection, and potential fire related exposure to which the wrap may be exposed (direct flame impingement, HEAF, etc.).	This F&O has been closed. Assessment of mechanical damage to fire wrap is not properly documented and multiple cases were found in the Fire Scenario Report 0493060006.004, where wrap is credited in HEAF scenarios. Therefore the F&O is still open. This F&O is superseded by F&O 10-4 from the focused peer review.
3-10	The PTN Fire PRA used much lower HRR for the evaluation of transient fires. The HRR for transient fires is based on the NUREG/CR-6850 HRR for electric motor fires (See F&O 3-10). The Fire Scenario Report (Report 0493060006.004) indicates that a transient NPP transient fire is better represented by a temporary cable installation, which includes an ignition source. Based on this the Fire Scenario Report indicates that the electric motor HRR is used to describe transient fires. A review of the EPRI Fire DB of transient fires indicated the following: a) events in the DB indicated that the fire was either above 75 kw or could have been above 75 kw, if not suppressed. For example, one transient fire resulted in an automatic suppression system actuation, which was likely above 75 kw due to the sprinkler head being above 160 F as a result of the fire, b) a recent event at one of the peer review team members plant was above 75 kw. As a result of this review, and discussion amongst the peer review team members, the HRR for	FSS-C4 FSS-D6 FSS-G1	Transient fire evaluations conducted as described in the Fire Scenario Report result in screening fire damage to targets due to the lower HRR which is believed to be non-conservative for developed fires involving ordinary combustible fuel packages such as a trash can or trash bag. Use the NUREG/CR-6850 HRR for transient fires or provide alternate justification for an area-specific HRR based on the limiting fire that could occur within the area.	This F&O has been closed. PTN has used 317kW throughout, except where strict transient controls will be implemented, 69kW used, which is consistent with EPRI methods. F&O is partially closed, however, suggestion F&O 10-10 from the focused peer review remains to update documentation to reflect implementation item as necessary, and to document that assumptions in the FPRA are within the limitations and conditions of the EPRI method.

	transient fires does not appear to be substantiated for the PTN FPRA. (This F&O originated from SR FSS-D6)			
3-11	The screening criteria is defined in the Turkey Point Hot Gas Layer and Multi-Compartment Analysis, (Report H0493060006.006) methodology. Compartments that don't screen are retained for further analysis. A concern identified with the screening criteria involves the use of a standard fire scenario for each analysis rather than determining the most challenging fire scenario inherent to the analyzed compartment. This approach potentially masks the potential for forming an HGL in the exposing compartment. (This F&O originated from SR FSS-G2)	FSS-G2	NUREG 6850, Section 11.5.4.3 Step 3.c.1 recommends development of a conservative HRR based on a combination of ignition source and secondary combustibles that produce the highest HRR. This recommendation is provided to ensure that a conservative/realistic HRR is used to determine the potential for HGL formation. Use of a non-conservative HRR may lead to underestimation of the potential for HGL formation and accordingly spread of potentially damaging hot gas to adjacent compartments. A realistic HRR should be developed for each fire compartment to ensure that the potential for formation of an HGL is appropriately assessed on a compartment by compartment basis.	This F&O has been closed. This F&O addressed the use of a standard fire scenario for compartment screening rather than the most challenging scenario. This approach is still used and F&O 9-5 from the focused peer review was generated to re-document this concern..
3-13	Review of fire modeling in enclosed compartments does not appear to accurately consider the addition of HRR from secondary combustibles. For instance a switchgear cubicle fire located in fire zone 71 is estimated to damage targets above the cubicle 4' horizontally and 7' vertically. The switchgear cubicles are vented at the top so a fire in these cabinets can be expected to spread to the cable trays above. The cable trays would be expected to ignite within the plume of the cabinet which is estimated to be at least 3' wide. Given that an initial width of 3' is reasonable, fire spread and additional HRR due to the resulting cable tray fire only assumes 1' of fire spread along the cable tray. Compared to the NUREG/CR-6850 guidance for flame spread along PVC cable (flame spread = 0.9 mm/sec) the estimation of HRR for these typical scenarios is nonconservative. Realistic estimation of the scenario HRR is necessary to ensure the full impact of the fire on exposed targets is presented and that the effects of a damaging HGL may also be estimated. (This F&O originated from SR FSS-C1)	FSS-C1 FSS-G1	Systematic application of generic fire modeling results to define target damage when fire spread should be postulated. In situations where the generic fire modeling treatments demonstrate fire spread to secondary combustibles, the scenario should assume full area damage. These scenarios are candidates for detailed fire modeling that would be used to show fire damage to PRA targets and the time to damage allowing credit for non-suppression probability.	This F&O has been closed. Review of fire modeling in single compartments does not consider the addition of HRR from secondary combustibles. It is acknowledged that secondary combustibles were considered for the MCA/HGL evaluation. FP&L has stated that walkdowns are in progress to include fire spread to cable trays and incorporate this into the fire scenarios. This issue therefore remains open and the F&O is superseded by F&O 10-16 from the focused peer review.
3-14	According to the generic limitations contained in Attachment B of the Fire Scenario Report (Report 0493060006.004) the generic fire modeling treatments do not account for the effects of hot gas layer (HGL) on the correlations presented. The limitation indicates that because HGL is not considered that these correlations should not be used in enclosed areas with small volumes where a significant HGL thickness may form. Because this relationship is not considered plume temperatures may be underestimated because it is assumed that ambient temperature air is being entrained into the plume, resulting in cooler plume temperatures, rather than heated air from the hot gas layer. Entrainment of heated air into the fire plume results in higher damage heights because the plume remains hotter at higher	FSS-C1 FSS-G1	Systematic application of generic fire modeling results to define target damage when fire spread should be postulated. The generic treatments used in relatively small rooms should be scrutinized to ensure that any HGL interaction is considered and accounted for if found to be significant.	This F&O has been closed. Subsequent to the 2010 review, FP&L states that "The impact of a hot gas layer on the zone of influence is evaluated for all fire zones/scenarios in the MCA/HGL evaluation." A review of this evaluation confirms that HGL effects on ZOI were in fact considered for the generic treatments; however, there is not sufficient documentation in Attachment A to the Fire Scenario Report 0493060006.004 to determine which zone of influence was applied to which scenario, and whether it was applied correctly to consider the effects of HGL. Issue is considered still open and is superseded byF&O 10-17 from the

	elevations. (This F&O originated from SR FSS-C1)			focused peer review.
4-4	There does not appear to be a document recording the proper resolution of deficiencies from the previous peer review of the internal events model. In addition, a peer review or gap assessment of the major changes since the previous peer review of the internal events model does not appear to exist (a gap database was provided, but with no supporting documentation). Finally, a gap assessment of the PRA standard changes from RG 1.200 Rev. 1 to Rev. 2 does not appear to exist. (This F&O originated from SR PRM-B2)	PRM-B2	SR PRM-B2 and NEI 07-12 require proper disposition of the deficiencies from the previous peer review that may adversely affect the accuracy of the Fire PRA model. CDF has been significantly reduced (more than an order of magnitude) since the last internal events peer review. Additionally, methodology changes have occurred, including use of the HRA calculator and use of a new CCF model. Resolve all significant deficiencies from the previous peer review that may adversely affect the accuracy of the Fire PRA model results, if not already completed, and document these dispositions. Provide a gap assessment or new peer review on the internal events PRA, latest revision.	This F&O has been closed. A draft Attachment U for the NFPA 805 License Amendment Request was provided to the review team. However, this document did not specifically address the question of whether any of the adopted F&O dispositions for the internal events PRA might have an impact on the fire PRA. The need for additional clarification of this is documented in F&O 8-3 from the focused peer review which supersedes this F&O.
4-5	Documentation of the changes in the Internal Events PRA model to develop the Fire PRA model are primarily provided in Section 4 and Appendix D of the Fire PRA Component and Cable Selection Report (Report 0493060006.001). However, no additional, detailed documentation of the changes is provided. Based on a review of the documentation in the tables in these sections (e.g., Tables 4.1-2, D-1, D-2, D-3, etc.), SRs for IE-D, AS-C and SY-C, etc. are not met. Since the level of analysis and documentation for Fire PRA model is expected to be similar to that for internal events, the documentation in the above sections and tables does not meet what is expected for this requirement. Let's look at some examples: on Table D-3, page D-57, there appears to be a number of changes to 'correct' logic. The original logic is included in the system notebooks (e.g., DC power notebook). However, since this is just correction here, the internal events PRA is not updated. Additionally, without proper development, it is difficult to determine if the change is accurate. (This F&O originated from SR PRM-C1)	PRM-C1	The level of documentation provided in Section 4 and Appendix D of the Fire PRA Component and Cable Selection Report (Report 0493060006.001) does not meet the SRs for IE-D, AS-C and SY-C. Include additional documentation to the level satisfying the SRs for IE-D, AS-C and SY-C.	This F&O has been closed. Changes made to the model to incorporate fire impacts are addressed in the PTN PSA Model Update Calculation, PTN-BFJR-00-001, Revision 9. The documentation provided is consistent with the process normally used for PSA model updates and is considered sufficient to disposition this F&O.
4-6	The Fire Scenario Report (0493060006.004) Appendices A, D, and E, and the FRANC model document the equipment failure modes for each fire scenario. However, circuit failure modes associated with failures of the required cables were not identified or documented. Relevant circuit failure modes are necessary for the assessment of circuit failure (e.g., hot short) probabilities. For most components, there is no differentiation in the FPRA between failure modes that can result due to failure of each cable/circuit. As a result, the fire scenarios assume each failure mode would occur from damage to all cables identified in the SSA. In order to refine the fire scenario under CF tasks, the circuit failure would need to be provided for each risk relevant circuit. (This F&O originated from SR FSS-A2)	FSS-A2	Circuit failure modes are necessary for the assessment of circuit failure (e.g., hot short) probabilities and required for meeting SR FSS-A2. The method used in the PTN FPRA can significantly over estimate the likelihood of the circuits causing such particular failure modes as spurious operation. Document circuit failure modes for the required cables for each fire scenario evaluated.	This F&O has been closed. Closed out based on evidence that circuit failure modes were used as evidenced the application of spurious operation probabilities via the altered events table.

4-8	The process of defining fire scenarios with the source/target combinations and its FRANC implementation process are such that the risk contribution of each risk-relevant ignition source was characterized. The evaluation and results are documented in the fire scenario report and the FRANC model. However, the analysis does not appear to always differentiate between targets (e.g., cables in different cable trays). For a number of the top 10 scenarios (i.e., full zone burn-out scenarios), it appears all fires damage all equipment and all cables in all trays (without specific knowledge of where each of the targets are located, for example, in the trays). Basically, the process for developing detailed scenarios for all significant fire compartments has not been completed in the FPRA for CDF and LERF. For example, see scenarios in 096, 019 and 020 (base case CDF scenarios). (This F&O originated from SR FSS-A5)	FSS-A5	Issue with potentially significant impact. For risk-significant fire compartments, develop additional fire scenarios such that specific targets are determined based on the location of each target (e.g., affected tray) relative to the ignition sources in the room.	This F&O has been closed. Additional fire modeling analysis for the rooms and scenarios identified has been performed and is documented in Fire Scenario Report 0493060006.004.
4-9	It appears that the application of severity factors is inconsistent between the FRANC model and those listed in Appendix A of the Fire Scenario Report (0493060006.004). In the FRANC model, severity factor is only used for transient fires and as the split fraction between severe and non-severe MCC fires. For transient fires, it appears that the severity factor is used in the sense of a location factor associated with the placement of the transient fuel. In Appendix A of the fire scenario report, severity factor is also used for oil fire, pump fire, and electrical cabinet fire, in addition to transient fires and MCC fires. Further, the values of the severity factor used in the FRANC model and Appendix A of the fire scenario report do not match. The bases for neither were documented for each individual scenario. Also, the severity factor values used for the oil and pump fires do not appear to be consistent with the tabulated values given in Table 5-1 of the fire scenario report. It appears that severity factors or non-suppression factors can be applied to many more detailed scenarios in the FRANC model to make the estimate of the detailed scenario risk more realistic. In addition, the scenario ignition frequencies listed in Appendix A do not appear to be consistent with those used in the FRANC model. (This F&O originated from SR FSS-C4)	FSS-C4 FSS-G1	This appears to be a systematic issue. However, this may also be just a configuration control and/or documentation issue. Severity factor should be applied to all applicable scenarios to derive realistic result. Reconcile the differences between the FRANC model and the fire scenario report. Apply the severity factor or non-suppression factor to all applicable scenarios in a manner consistent with the methodology and data discussed in the fire scenario report.	This F&O has been closed. Appendix A to Fire Scenario Report 0493060006.004 has been updated to reconcile inconsistent use of severity factors between the Fire Scenario Report and the FRANC model. Appendix H has been added to the Fire Scenario Report to document basis for ignition frequencies.
4-10	The HRR used for fire modeling of the zone of influence is based on motor fires, which is substantially lower than the NUREG/CR-6850 recommended HRR of 317 kW. As a result, the use of severity factor could potentially be double counting the lowered HRR for transient fires (note that even when the severity factor is used as the location/placement factor for transient fires, it is dependent on the HRR in terms of the zone of influence), if the severity factor development is based on the NUREG/CR-6850 HRR for transient fires. The severity factor for transient fires discussed in Section	FSS-C4 FSS-D3 FSS-D5 FSS-E2 FSS-G1	Severity factor is used extensively in the Fire PRA. Use the NUREG/CR-6850 HRR for transient fires, or develop an accepted industry HRR approach (presently being discussed by EPRI). Develop transient fire severity factors based on the likely HRR and location of overhead cables or location of equipment. For example, if cable is 7 feet overhead, the severity factor would be based on the minimum HRR that would damage the cable at that distance. Additionally, the growth time can be used in determining non-suppression time for generic cases, based on the latest FAQ 52. Finally, it is	This F&O has been closed. Subsequent to the 2010 peer review, transient fire modeling in Fire Scenario Report 0493060006.004 has been revised to include consideration of a 317kW fire HRR, and the severity factor eliminated. Therefore the 1st part of this F&O is closed. In lieu of using a severity factor that is independent of other factors, and that bounds plant specific conditions, a panel split fraction was used, which is considered an unreviewed analysis method,

	8.4 of the fire scenario report (0493060006.004) does not provide sufficient justification for a generic transient fire severity factor. In addition, the severity factor derived from an analysis of the number of fire events includes non-suppression results, and would therefore not be independent of any non-suppression probabilities applied later. FSS-C4 requires severity factors to be independent of other factors. It is noted, however, that the implementation of the severity factor in the PTN FRANC model did not involve the application of both severity factor and non-suppression factor in the same scenario. Fire severity factor as discussed in Section 7.1.2 for low voltage electrical cabinets is not developed or applied consistently with the NUREG/CR-6850 methods. This is developed from a supplemental report (ERIN report, Supplemental Fire PRA Methods). Additionally, fire propagation outside of the electrical cabinets is also dependent on the non-suppression probability. Therefore, some dependency exists in this data if used in conjunction with a non-suppression factor. The numbers listed in 7.1.2 for electrical cabinets were derived using the total number of cabinet fires in the denominator, rather than the number of fires of the specific panel type. Due to this incorrect derivation of the conditional probabilities for fire propagation outside of the cabinets, the conditional probabilities thus developed (and applied in the FRANC model for low voltage cabinets) could potentially be low by an order of magnitude (non-conservative). For both the transient fires and low voltage cabinet fires, the severity factors are basically developed using fire events data from the EPRI report. Given the fire data duration and damage is a result of multiple factors (growth, suppression, severity, location, etc.), and given the fire data often does not have sufficient information to make a reasonable determination of either the fire size or whether a fire propagated outside the cabinet, the peer review team determined that the use of fire events data for developing the above severity factors is not appropriate. (This F&O originated from SR FSS-C4)		recommended that the current conditional probabilities in 7.1.2 table for electrical cabinets should not be used in the FPRA.	therefore, the issue is still open. This F&O is superseded by F&O 10-3 from the focused peer review.
4-11	The argument in Section 8.1 of the fire scenario report about the application of the probability of an ignition source being located within an area around the target may not be correct. This is because the application of a location factor for the transient fuel to the fire ignition frequency has already accounted for the probability of the target being within the influence zone of a fire. As such, ignition is a given condition. With the apportioned frequency, the target must be located within the impact area around the transient fuel and an ignition source must be located within an area in the vicinity to ignite the transient fuel. (This F&O originated from SR FSS-C4)	FSS-C4 FSS-D3 FSS-D5 FSS-G1 FSS-H6	The location/severity factor applied should have already accounted for the probability that the target is located within the impact area around the transient fuel and an ignition source is located within an area in the vicinity to ignite the transient fuel. Revise Section 8.1 and do not apply separate location factors for transient fuel and ignition source simultaneously in any fire scenario.	This F&O has been closed. Transient fire modeling and section 8.1 to Fire Scenario Report 0493060006.004 has been revised to remove the separate transient fire location factors, and the approach is now consistent with industry approved methods.

4-12	Except for the MCR fire scenarios, no scenario-specific fire modeling has been performed to provide a mean value of, and statistical representation of, the uncertainty intervals for the parameters used for modeling the compartments with significant fire risk contributions. Therefore, for compartments other than MCR, only the results of conservative, generic fire modeling developed in the Generic Fire Modeling Treatments report were applied to the analysis of fire scenarios. (This F&O originated from SR FSS-E3)	FSS-E3 FSS-H9 UNC-A2	Mean values and uncertainty intervals for the parameters used for modeling the significant fire scenarios have not been provided. Consider developing mean values and uncertainty intervals for the parameters used for modeling the significant fire scenarios.	This F&O has been closed. Parametric uncertainty analysis has been performed for Unit 4 (to be performed on Unit 3 using same methodology)
4-13	Uncertainties associated with cases where cable routing has been assumed (e.g., the EXCLUDEDEVENTS table has assumed that certain cables are not routed through selected areas based on walkdown or engineering judgment) have not been investigated with a documented basis. (This F&O originated from SR FSS-E4)	FSS-E4 FSS-H9 UNC-A2	The required uncertainty has not been evaluated / performed. Consider investigating the uncertainties associated with cases with assumed cable routing.	This F&O has been closed. Cable with previously unknown routing which are credited in the FPRA have been traced via RFI-0279.
4-14	The Hot Gas Layer and Multi-Compartment Analysis report (H0493060006.006, Revision 0) performed a screening evaluation of the need for hot gas layer and multi-compartment analysis, and identified scenarios/zones that warrant further evaluations. No detailed multi-compartment analysis is completed (still in progress) in this report, and no discussion on multi-compartment fire scenarios' risk contribution is provided. (This F&O originated from SR FSS-G6)	FSS-G3 FSS-G6	Multi-compartment fire scenarios' risk significance is not yet evaluated. Complete the detailed multi-compartment analysis and add the discussion on multi-compartment fire scenarios' risk contribution.	This F&O has been closed. This F&O concerned completion of the MCA. MCA analysis was completed
4-15	Treatment for transient fire damage to targets is measured from the compartment floor rather than the height of the transient fuel package that is typically considered. PTN FSS report section 8.2 states, "Cable trays (or the lowest tray within a stack of trays) that were at least 5.8' off the floor were considered beyond the zone of influence of the transient fire for nonqualified cables." This apparently was based on the lowered HRR values (See F&O 3-10) used for the transient fires, as well as the transient fire being located at the floor. This may result in the reperformance of transient fire walkdown if the transient fire HRR values need to be updated. Discussion with FP&L following the onsite review provided some basis for the damage height (indicating that transient fires above the floor will have an overall lower average surface HRR). However, the supplemental discussion was still considered inconsistent with past events and existing guidance on analysis of transient fires. (This F&O originated from SR FSS-H6)	FSS-D6 FSS-G1 FSS-H6	Significant modeling issues resulting in many transient fire scenarios being screened during detailed scenario analysis. The results of the FPRA are therefore potentially non-conservative for the analyzed detailed scenarios. Transient fire evaluations conducted as described in the Fire Scenario Report result in screening fire damage to targets that are located > 5.8' above the floor which is believed to be non-conservative for developed fires involving ordinary combustible fuel packages such as a trash can or trash bag. In response to this concern it was pointed out that the thermal plume component relies on empirical relationships between the source strength and the distance between the virtual origin of the fire and the target. The fire plume begins to entrain air at the lowest point of burning, which defines the base of the fire; normally at the floor. However this argument ignores the potential that a fire could begin burning at the top of a fuel package thus elevating its base. At a minimum, during the initial period of burning, damage temperatures generated by the fire would likewise be elevated. Over time the base of the fire may change due to collapse of the fuel package or burning away of the fuel, however the empirical model presented did not develop these ideas as a reason for assuming that the base of the fire is at the floor for its entire duration. The	This F&O has been closed. Treatment for transient fire damage to targets is measured from the compartment floor rather than the height of the transient fuel package that is typically considered. Discussion with FP&L during the review provided some basis for the damage height (indicating that transient fires above the floor will have an overall lower average surface HRR). However, the supplemental discussion was still considered inconsistent with past events and existing guidance on analysis of transient fires, and could lead to non-conservative estimates of transient fire damage to targets. This F&O is superseded by new F&O 10-6 from the focused peer review.

			transient fires should be considered to be above the floor level in the analysis.	
4-17	Per summary report, Task 9 is fulfilled with the NISYS SSD database, "PTN NFPA 805 Database.mdb". This database has been significantly expanded for the NFPA 805 tasks. A sample circuit analysis worksheet (e.g., for component 20ASB/G3) has signatures at the bottom, which were not populated yet. The NISYS circuit analysis is an Appendix R type circuit analysis and does not identify the circuit failure modes and address likelihood of failure. Failures of the required cables identified are assumed to have a probability of 1.0 unless specifically modified in the ALTEREDEVENTS table of the FRANC model. The treatment of the circuit analysis seems to be bounding (i.e., the likelihood was not part of the analysis). Although Appendix D of the fire scenario report states the bases for the altered FRANC event probabilities, it seems that there is no linking between the altered probabilities and the circuit analysis package. The majority of the altered events are based on operator manual actions while some based on the simple spurious actuation probabilities from NUREG/CR-6850, which were based on specific evaluation (with no basis provided in the FRANC database), but do not directly linked to any specific circuit analysis worksheet. Since the "basis" column of the Altered event table in the FSS report does not appear to include sufficient documentation to allow review/peer review of the results and the NISYS database does not include the analysis, the analysis (not the results) has not been documented. The evaluation and documentation of the review of the fire-induced circuit failure modes and the assignment of the appropriate industry-wide generic values to their conditional failure probabilities for risk-significant contributors based on the specific circuit configuration under consideration should be included in the circuit failure report for Tasks 9 and 10. (This F&O originated from SR CF-B1)	CF-B1	Incomplete evaluation and document for circuit failure. The NISYS DB can include identification of when spurious operation may occur, but does not provide the circuit analysis or circuit failure probability analysis needed to support the FPRA. Provide a documented basis, and detailed circuit analysis for any spurious operation probability used in the FPRA per Tasks 9 and 10 of NUREG/CR-6850 (or equivalent).	This F&O has been resolved. Circuit failure probability was considered for high risk scenarios and only in cases where doing so would result in a reduction in total risk. Additional details with respect to circuit configuration and raceway type have been added to the altered events table.
4-19	The new fire-specific safe shutdown actions identified and incorporated into the PTN Fire PRA have not been defined (even for the risk-significant actions) by specifying (a) accident sequence specific timing of cues, and time window for successful completion (b) accident sequence specific procedural guidance (e.g., AOPs, and EOPs) (c) the availability of cues and other indications for detection and evaluation errors (d) the specific high level tasks (e.g., train level) required to achieve the goal of the response, or the complexity of the response. (This F&O originated from SR HR-F2)	HRA-B2 HRA-B3 HR-F2	Risk significant human actions should be defined in accordance with SR HR-F2. Identify the risk-significant new fire-specific safe shutdown actions and define these actions in accordance with SR HR-F2 and HRA-B3.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is superseded by new F&O 7-6 from the focused peer review.

4-23	The time available and time required to complete actions were not evaluated for the new, risk significant fire-related safe shutdown actions. PTN FPRA HRA report Tables A-1 and A-2 include evaluation of the time available to complete actions. However, the point in time at which operators are expected to receive relevant indications are not evaluated. (This F&O originated from SR HR-G4)	HRA-C1 HR-G5 SF-A5	Systematic issue. The evaluation of the time available and time required to complete the risk-significant actions is required. Evaluate time available to complete the risk-significant fire-related safe shutdown actions.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is superseded by new F&O 7-6 from the focused peer review.
4-24	It appears that the reasonableness of risk-significant, post-initiator HEPs relative to each other was not yet reviewed in the scenario context, plant history, procedures, operational practices, and experience. (This F&O originated from SR HR-G6)	FQ-C1 HRA-C1 HR-G6 QU-C1	The reasonableness of risk-significant, post-initiator HEPs relative to each other should be reviewed and checked in the scenario context, plant history, procedures, operational practices, and experience. Review the reasonableness of risk-significant, post-initiator HEPs relative to each other in the scenario context, plant history, procedures, operational practices, and experience.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is superseded by new F&O 7-6 from the focused peer review.
4-25	Uncertainty characterization of the HEPs developed for the Fire PRA was not provided. (This F&O originated from SR HR-G8)	HRA-C1 HR-G8 QU-E4 UNC-A1	HEP uncertainty characterization is needed for the evaluation of uncertainty in the overall risk results. Develop uncertainty characterization of the HEPs used in the Fire PRA (especially for those risk-significant HFEs).	This F&O has been closed. Draft parametric uncertainty analysis for an earlier version of the model was provided. This includes application of uncertainty bounds for the screening HEPs applied through altered events. There is also a brief discussion of sources of model uncertainty in Appendix D of the TURKEY POINT NUCLEAR PLANT FPRA SUMMARY REPORT NUREG/CR-6850 TASK 16 (0493060006.005). The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is superseded by new F&O 7-6 from the focused peer review.
5-3	Based on a review of the Turkey Point FPRA Component and Cable Selection Report 049306006.001 Rev. 1, the documentation is sufficient to support the supporting requirements. The documentation issues identified in the F&Os generally include suggestions to clarify information in the calculation to accurately reflect the process followed in the PRA. ES-D1-01 concerns a finding related to the MSO attachment. This attachment needs to be updated to clearly document how each MSO was addressed in the model in	ES-D1 PRM-B3 PRM-B4	The report identifies multiple cases with confirm or investigation required. Some of the resolutions were documented as addressed, but there some cases with no documentation of resolution or justification can be found. The PRA documentation does not clearly show the resolution of the MSO items as required by the standard. This is considered a documentation concern because the resolution of these items can be found in the PRA model. Suggestion to update Appendix A with a clear resolution of all MSO identified open	This F&O has been closed. Additional MSO Expert Panel sessions were conducted and are documented in the Expert Panel for Addressing Multiple Spurious Operations Report (0027-0003-003-001, Revision 1). The document now includes a description of how the MSO impact is incorporated in the model. In addition, PTN PSA Model Update Calculation, PTN-BFJR-00-001, Revision 9 specifically references those changes

	accordance with the actions from the MSO expert panel. Another example: scenario 45 involving (diesel overload) has a note to complete the on-going evaluation; scenario 34 (loss of SG inventory) has a note to verify total flow rates; ensure adequate Fire PRA documentation. Another example is scenario 7, 'Normal letdown fails to isolate and inventory is lost to the pressurizer relief tank (PRT)', which has an open action to revisit PRA model structure for letdown isolation. The FPRA documentation doesn't discuss the disposition for this scenario and the FPRA doesn't model it. (This F&O originated from SR ES-D1)		items. Provide documentation on open items from the MSO expert panel. Include modeling in the FPRA of any known scenarios that are found to be an issue.	that are related to an MSO scenario. This is considered sufficient to disposition this F&O.
5-5	Turkey Point FPRA Human Failure Evaluation Report 0493060006.002, page 6 states that a simulator review was performed to identify instrumentation that should be explicitly modeled in the FPRA, including "Identify any alarms or indications that would lead operators to take immediate control actions without further verification." Appendix C of the same report, page C-3 has a section asking which annunciator tiles cause an operator to take immediate action. The response is "Operators will confirm the signal with an alternate indicator before taking any action." This answer is not specific, and may not always be procedurally correct. In a typical NPP, each annunciator sends the operators to some alarm response procedure. The alarm response procedure will typically require confirmation using an alternate indicator; however, this is not always the case. (This F&O originated from SR ES-A2)	ES-A2 ES-C2 HRA-A3 HRA-B4	Any alarm procedure that does not require alternate confirmation will need to be reviewed and either dispositioned or assessed for equipment to be added to the FPRA component list. There does not appear to be an extensive review of these alarm response procedures for either identification of instrumentation failures leading to a trip or causing an operator to shutdown plant equipment. Review alarm response procedures to address this issue and document appropriately. This aligns with the guidance in section 2.5.5 of NUREG/CR-6850.	This F&O has been closed. There has still not been a systematic evaluation of the potential for undesired operator actions as a result of a single instrument failure. However, since generic evaluations performed by the Pressurized Water Reactor Owners Group has addressed this issue for actions based on the EOPs, and the fact that the Turkey Point annunciator response procedures do not include immediate operator actions that have been problematic in other evaluations, this is not expected to be a significant issue for the PTN fire PRA. However, suggestion level F&O 7-5 from the focused peer review was generated to document the need to perform a systematic review to meet Capability Category II for this SR. F&O 5-5 is superseded by this new F&O.
5-11	Review of Turkey Point NISYS NFPA 805 Compliance Assessment Database within the Cable Routing and Respective Equipment table, it was noticed that the Spurious ESFAS signal "Spurious/ESFAS/Lacks/Analysis" have total of 56 respective components impacted. Unit 3 Train A SI signal from the Control Room "3MRASI/3C06/3QR43/006" have total of 29 respective components impacted, Unit 3 Train B SI signal from the Control Room "3MRBSI/3C06/3QR45/006" have total of 28 respective components impacted, Unit 4 also have similar components impacted. The concern is the potential mismatch between FPRA and the SSA component lists. (This F&O originated from SR ES-B1)	ES-B1	The deviation between the ESFAS components actuation and Control Room SI components actuation should be disposition and reconcile, to ensure Fire Safe Shutdown / Appendix R equipment are appropriately credited in the Fire PRA. Reconcile the FPRA component list with the SSA component list for equipment impacted by an SI signal	This F&O has been resolved. The circuit analysis process used for the project has been confirmed to be consistent with the latest industry guidance (NEI 00-01). In addition, the asymmetry was discussed with plant staff and confirmed to be reflective of the actual plant design and configuration.
5-13	Turkey Point FPRA Summary Report NUREG/CR-6850 Task 16 Report No. 049306006.005 Rev. 1 Tables A-1, A-2, B-1 and B-2 documented the Units 3 & 4 Fire PRA quantification Results for both CDF and LERF for all fire scenarios that were quantified. Scenario 096-A was randomly picked review for both Units 3 & 4. The CDF/LERF results are consistent between the Summary Report and Zone Scenarios in database files, Unit 3 CDF	FQ-A3	It appears that there is inconsistent basic event mapping between the database files. A sensitivity run was performed by copying the U4 events to the U3 tables, and re-evaluated U3 CDF. The results are the top scenario in 96 dropped from 4.5E-05 to 1E-06. Based on this, the error appears to be significant. Need to ensure that the altered events table is correctly developed for both U3 and U4 for the CDF and LERF	This F&O has been resolved. The identified data differences were reviewed and confirmed to be reflective of the design and layout of the units. Additional comparison of the quantification results between the two units was also performed to ensure that any significant differences in results are consistent with the actual unit differences. Various

	"PTNFIRE_W_LERF_MH_ESF.mdb", Unit 3 LERF "PTNFIRE_W_LERF_MH_ESF.mdb", Unit 4 CDF "U4PTNFIRE_W_LERF_MH_ESF.mdb", and Unit 4 LERF U4PTNFIRE_W_LERF_MH_ESF.mdb". However, reviewing the Altered Events table in each database files shows inconsistent basic events impacted between Unit 3 and 4. Unit 3 have no basic event impacted, while Unit 4 have 9 basic events listed. (This F&O originated from SR FQ-A3)		quantification.	asymmetries in the plant layout were identified.
5-16	Review of PTN Tasks 8 and 11 Report 0493060006.004, Rev. 1. Page 23, Section 7.5.2, states "no hydrogen fires other than turbine/generator have been postulated. The basis appears to be that they use excess flow check valves to limit H2 release. Question was asked during the review, the response said "The small quantify of hydrogen downstream of the check valves and its potential leakage will result in small accumulations of hydrogen and are unlikely to result in combustible concentrations of hydrogen in any area of the plant". However, further review of PTN NFPA 805 Fire Ignition Frequency Report PTN-PSA-7.01, Rev. 2, Fire Compartment Ignition Source Data Sheet (ISDS) for compartment 082 "Unit 4 Auxiliary Transformer Area" indicate that BIN 19 - Misc. Hydrogen Fires was identified in this area. For example, H2 feed to the VCT should be looked at. (This F&O originated from SR FSS-A1)	FQ-A3 FSS-A1	The discussion between the two reports are inconsistent regarding to misc hydrogen fire. Incorrect apportioning the fire frequency and define appropriate fire scenarios could have significant impact to the CDF and LERF results. Address Miscellaneous H2 fires in the FPRA, identifying other compartments containing hydrogen piping.	This F&O has been closed. The incorporation of analysis for Hydrogen fire scenarios for Fire Compartments 45, 55, 82, and 87 was still in progress. This F&O is superseded by new F&O 10-2 from the focused peer review.
6-1	No documentation is provided of a comprehensive review of fire impacts to plant equipment for unique initiating events. The MSO list includes combinations of spurious operation components. However, this review does not include consideration of combinations of fire-induced failures that can lead to an initiating event. Additionally, there did not appear to be a review of screened initiating events from the internal events PRA, other than ISLOCA pathways. (This F&O originated from SR ES-A3)	ES-A3 ES-A4 ES-D1 PRM-B3 PRM-B4	Systemic issue Perform a review of fire scenario equipment impacts to identify fire-specific initiating events. Examine groups components that can be disabled by a single fire and include the potential for a single spurious event. Examine these equipment impacts in terms of plant response, timing, success criteria, and the effects on the operability and performance of operators and mitigating systems. For each scenario, identify a new fire-specific initiating event if no existing initiating event bounds or adequately represents the equipment impacts.	This F&O has been closed. The potential for multiple equipment impacts resulting in a more severe event than the individual impacts is addressed in the modeling. For example, if a single fire could cause spurious operation of multiple valves that would individually result in a small-small LOCA, these are treated as a small LOCA. However, there is still additional documentation is documented in suggestion level F&O 8-4 from the focused peer review which supersedes this F&O.
6-3	For the FPRA, no accident sequences were identified beyond those modeled by the internal events PRA. The FPRA accident sequence accident progression, success criteria and timing are therefore based on the internal events PRA. Consideration should be given, however, to success criteria and timing specific to the FPRA. For example, no evaluation is made of the timing associated with RWST drain down. Also, RWST drain down may require sump recirculation, which is not presently represented in the non-LOCA transient event tree accident sequences. This step has not been performed and finding is made to include such considerations in the FPRA	AS-A1 AS-A9 LE-A1 LE-A2 PRM-B14 PRM-B15 PRM-B5 PRM-B7	Required step not performed. Perform a review of FPRA scenarios to ensure that the existing event tree structures accurately model the specific FPRA initiating events, including considerations of timing, plant response, and human interactions.	This F&O has been closed. Subsequent to the 2010 peer review the updated LERF model has been completed and is documented in FPL Calculation PTN-BJFR-99-010, Revision 1. However, there is still a need to address fire-specific impacts on the accident progression sequences. This is superseded by F&O 7-3 from the focused peer review.

	development and documentation. Also, when any new FPRA initiating events are identified as part of resolving ES F&Os, consideration will need to be made of the accident sequence accident progression and timing associated with any new accident sequences. (This F&O originated from SR PRM-B7)			
6-4	No FPRA modeling appears to have been made to address the actions directed by the fire safe shutdown procedures that deviate from the actions directed by the EOPs. (This F&O originated from SR PRM-B6)	AS-A1 AS-A10 AS-A4 AS-A5 HRA-A1 HRA-A2 HR-E1 HR-E2 PRM-B5 PRM-B6	Step not performed. Potentially significant impact on FPRA accident sequences and results. Modeling of the following procedural responses to a fire may be needed: 1) equipment is disabled to preclude spurious actuations; 2) human actions to isolate unprotected equipment; 3) human actions to manually operate protected equipment. Some of the above human actions could also induce new sequences not traditionally covered in the Internal Events PRA. New sequences to account for these effects may also need to be incorporated into the Fire PRA Model.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to consider fire-specific impacts on accident sequences and success criteria and to provide appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is documented in new F&O 8-1 which supersedes this F&O.
6-9	The parametric uncertainty analysis as discussed in QU-E3 (estimate of uncertainty intervals, etc.) is not performed. Also, the "state-of- knowledge" correlation between fire-specific event probabilities (e.g., suppression system unavailabilities, fire ignition frequencies, hot short conditional probabilities, etc.) hasn't yet been applied. (This F&O originated from SR QU-A3)	FQ-A4 QU-A3	Step not performed. Perform the FPRA uncertainty analysis, including estimates of uncertainty bounds, per the requirements of QU-A and QU-E. When performing parametric uncertainty calculations, ensure uncertainty intervals for event probabilities utilized by the FPRA are correlated when significant.	This F&O has been resolved., Parametric uncertainty has been performed for CDF and LERF for each unit's FPRA.
6-10	The altered events table in the FSS report includes several instances where a single basic event combines a hot short spurious operation likelihood with an HEP to recover the spurious operation. For example, ORZR30455C represents a combination of spurious opening of a PORV and operator human error probability to close the PORV. Supporting requirement FQ-A1 addresses the need to translate specific failure modes into basic events. Embedding an HEP with a spurious operation likelihood bypasses this requirement, and this approach is not consistent with the level of detail modeled elsewhere in the PRA. Also, the approach prevents the ability to address the state of knowledge correlation. (This F&O originated from SR FQ-A1)	FQ-A1 FQ-A4 HRA-E1 HR-I1 HR-I2 QU-A3	This approach is not consistent with the level of detail modeled elsewhere in the PRA. Translate specific failure modes into basic events and avoid combining disparate failure modes into combined basic events.	This F&O has been resolved. The methodology and the analysis has been updated to eliminate the use of this approach. The use of altered events for spurious probability is used only as required and a singular value.
6-11	The FPRA models through the Altered events table several 0.1 values for recovery actions that are not in the safe shutdown analysis and are not in the fire safe shutdown procedures. There are about 198 unique instances of such recovery actions. For example, event AHFPTRNAMAN, in area 63, is not included in operations procedure 0-ONOP-016.10. This human interaction modeling doesn't reflect the as built as operated plant, and no evaluation of feasibility is documented for these actions. Discussions with the FPRA	HRA-A1 HRA-A2 HRA-C1 HRA-D2 HRA-E1 HR-E1 HR-H2 HR-I1	The operator recovery actions were based on proposed new procedures instead of the existing ones, don't reflect the as built as operated plant and are not confirmed to be feasible. Ensure that all FPRA human failure events reflect the as-built, as-operated plant, and that they are proceduralized. Verify all credited actions, including those modeled in the HRA and those included in the altered events report, are included in the plant operational procedures.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and

	development team indicated that the post-fire operating procedures will be updated to incorporate the new recovery actions and feasibility will be evaluated at that time. The FPRA will need to be updated, as necessary to reflect the outcome of feasibility evaluations. (This F&O originated from SR HR-E1)	HR-I2		detailed HEP development for significant HFEs is documented in new F&O 7-6 from the focused peer review which supersedes this F&O.
6-12	Based on a review of the FPRA HRA Report, no talk-throughs or reviews appear to have been made with plant operations and training personnel of procedures and sequences of events to confirm that interpretation of the procedures by FPRA developers is consistent with plant observations and training. The report indicates that a simulator review was performed to identify instrumentation that should be explicitly modeled in the fire PRA as generally required to shutdown the plant or to perform credited operator actions, and a review agenda is provided in Appendix C, but no documentation of such a review is provided. (This F&O originated from SR HR-E3)	HRA-A1 HRA-A4 HRA-E1 HR-E3 HR-I1 HR-I2	Step not performed Perform talk-throughs or reviews of procedures and sequences of events with plant operations and training personnel to confirm that interpretation of the procedures is consistent with plant observations and training.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is documented in new F&O 7-6 from the focused peer review which supersedes this F&O.
6-13	No simulator observations or talk-throughs with operators have been performed to confirm the response models for fire scenarios modeled. (This F&O originated from SR HR-E4)	HRA-A4 HRA-E1 HR-E4 HR-I1 HR-I2	Step not performed Perform simulator observations or talk-throughs with operators to confirm the response models for fire scenarios modeled.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is documented in new F&O 7-6 from the focused peer review which supersedes this F&O.
6-15	The documentation of credited recoveries in the altered events table in many instances is vague (e.g., "Manual action to restore bus," and "failure probability of new inside control room HEP, required in less than 60 minutes (3AA, DC 205)"). The effort to evaluate these actions for inclusion in the SSA and feasibility would be facilitated by more detailed descriptions of the actions. PTN system model changes in the fire PRA models are summarized in Tables D-1, D-2 and D-3 of the Component and Cable Selection Report 0493060006.001, Revision 1. However, no additional documentation of the changes is provided. Requirements under SY-A and SY-B are not met. Repair of components that are spuriously operated or fire damaged is modeled using the altered events table of Attachment D, FRANC Altered Events Table, of the Fire Scenario Analysis Report 0493060006.004, Rev 1. However, the substitution does not include a verification that the actions are possible	HRA-E1 HR-I1 HR-I2 PRM-B9	Systematic Issue Perform analyses to verify operator recoveries and repairs credited in the Altered events Table in Appendix D are feasible, prior to crediting any recovery via manual operation of the equipment.	This F&O has been closed. The HRA for the fire PRA is still based on application of screening values using the ALTEREDEVENTS table in FRANC. This approach is to be maintained for the model used in the NFPA-805 License Amendment Request. The need to complete the Fire PRA HRA in a manner similar to the internal events HRA with appropriate levels of documentation for screening values retained and detailed HEP development for significant HFEs is documented in new F&O 7-6 from the focused peer review which supersedes this F&O.

	or feasible. (This F&O originated from SR HRA-E1)			
6-16	The HEP dependency evaluation produced several dependent HEPs on the order of 1E-11 to 1E-13 (for example, cases 62, 93, 110 and 96). NUREG-1792, "Good Practices for Implementing Human Reliability Analysis," recommends that the total combined probability of all the HFEs in the same accident sequence/cut set should not be less than a justified value. NUREG-1792 suggests that the value not be below ~1E-5, since it is typically hard to defend that other dependent failure modes that are not readily anticipated cannot occur. However, some industry PRAs are using a floor value of ~1E-06. (This F&O originated from SR HRA-C1)	FQ-C1 HRA-C1 QU-C2	Unreasonably low values assigned to dependent HEP combinations. Assign a floor for dependent HEP combinations using a justified minimum value.	This F&O has been closed. The issues raised in this F&O are based on good practices from NUREG-1792 which is not directly referenced in the ASME/ANS PRA Standard. As the basis for any SR. The practice being employed by FPL for the PTN fire PRA is consistent with that observed in recent internal events peer reviews using the HRA calculator. While there are still dependency issues to be addressed as documented in F&O 7-8 from the focused peer review, this F&O is considered to represent reviewer opinion and not compliance with the requirements of the standard.
6-20	The parametric uncertainty associated with conditional circuit failure probabilities are not evaluated and are not incorporated into the model. (This F&O originated from SR CF-A2)	CF-A2 UNC-A2	Step not performed. Develop uncertainty intervals for applied hot short probabilities and include them in the model.	This F&O has been resolved., Parametric uncertainty has been performed for CDF and LERF for each unit's FPRA.
7-1	A review of the quantification results for selected compartments involving fire-induced safety injection actuation and fire-induced opening of atmospheric dump valves was performed to verify that the modeling was consistent with the internal events PRA treatment of similar initiating events. This review revealed that duplicate cutsets were being introduced by the manner in which the new logic for capturing the fire-induced initiating events was linked into the fault tree. For example, a review of cutsets for zone 098-A showed that the top two cutsets were identical except that one used a version of the HFE for alignment of bleed and feed based on reactor trip occurring with SG low level and the other used a version of the HFE based on timing associated with trip with nominal SG level. Similar issues were identified in the cutsets for zone 091-ETL. As noted in the 2010 peer review in F&O 1-4, there are also inconsistencies in modeling of the fire-induced small LOCA when compared to the internal events small LOCA initiating event. PTN explains that this was due to circular logic issues, and a sensitivity case shows this to be a non-significant issue. However, it is not clear that the circular logic issue could not be resolved and that all potential impacts of the modeling approach taken are understood. (This F&O originated from SR PRM-A3)	PRM-A3 PRM-A4 PRM-B5 PRM-B9	The method in which the fire-induced spurious safety injection actuation and spurious opening of atmospheric dump valve initiating events were linked into the fault tree produces conservative results that could impact the determination of significant contributors to fire-induced risk. Review the quantification results for the fire-induced initiating events to verify that the results are consistent with the comparable internal events. Review the mapping of the fire-induced initiating event impacts to ensure that they are consistent with the comparable internal events initiator, that appropriate differences due to the considerations of the fire PRA are incorporated (e.g., application of bounding timing for HEPs to capture uncertainty in the sequence of fire-induced failures), or that deviations in the modeling are documented and justified. Review the application of the feed and bleed HFE to ensure the appropriate timing is used during the fire quantification. Since the MFW pumps are assumed failed for all fire areas, the most appropriate value may be the HEP based on timing assuming the trip occurs with low level in the SGs. Review treatment of any additional HEPs with event-specific timing assumptions to ensure that the appropriate values are used in the fire quantification.	This F&O has been resolved. A review of the model was performed and revision made to address and resolve the issue identified in the F&O. Additional reviews were performed as part of the overall results and cutset reviews and no additional instances were identified.
7-3	The current model uses the LERF model for the PTN revision 9 model (PTN-BJFR-99-010, Rev. 1) and maps appropriate equipment impacts into the system models used to model LERF. No new accident progressions beyond the onset of core damage were identified for the fire PRA. However, there is no documentation that a specific review	PRM-B14	It cannot be determined from the existing documentation that an assessment was performed to identify new accident progressions beyond the onset of core damage that would be applicable to the Fire PRA that were not addressed for LERF estimation in the Internal Events PRA. Document an assessment to determine if there are potential fire-induced	This F&O has been resolved. A review of the mapping of Level 1 sequences to the plant damage states in the LERF model was reviewed. No new accident progressions that required modification of the LERF model were

	of the accident progressions leading to LERF was conducted to identify whether new considerations should be addressed in the fire PRA. In addition, effects on PDS mapping due to fire-induced failures may not be appropriately captured. For example, RWST diversion of the RWST to the containment sump is modeled as a failure of HHSI which would normally go to a dry containment PDS. However, the actual PDS should be one for wet containment. While this is a late containment failure concern rather than a concern for LERF, there may be similar fire-induced failures that could affect the mapping of LERF accident progressions. (This F&O originated from SR PRM-B14)		LERF mechanisms not captured by the internal events accident progression models. If none are identified, document the basis of that conclusion.	identified.
7-6	The new fire-specific safe shutdown actions which are credited in the final Fire PRA will be proposed to be added to the plant fire response procedures. These human actions are included in the ALTEREDEVENTS table of the FRANC model using component basic events as surrogate. However, the safe shutdown actions modeled in the FPRA are not currently consistent with those specified in the plant fire response procedures, there is no documented assessment of the cues required to initiate the actions, no training has been provided to operators on the new fire-specific actions, no operator reviews or talk-throughs of the credited actions has been documented, and the applicable performance shaping factors have not been considered, including time available for the action. This F&O supersedes 2010 Peer Review F&Os 1-41, 2-6, 6-4 and 6-11. (This F&O originated from SR HRA-A2)	HR-E1 HR-E2 HR-E3 HR-E4 HR-H2 HR-I1 HR-I2 HR-I3 HRA-A2 HRA-A4 HRA-B2 HRA-B3 HRA-D2 HRA-E1 PRM-B11	Final post-fire safe shutdown actions have not been defined and appropriate procedures revised to include the actions to be credited in the Fire PRA. Complete the identification of new fire-specific safe shutdown actions which are credited in the final Fire PRA and evaluate and document the HEPs consistent with processes used for internal events HEPs. Include consideration of fire effects on the operator action, availability of cues, availability of time to complete the action, feasibility of the credited actions given a fire, and potential 92-18 impacts for both screening values and detailed HEP development. Also, complete operator reviews and/or talk-throughs when the procedure updates are completed to ensure that the interpretation of the actions is consistent with the operator's understanding and training. Finally, consider expanding the discussion of sources of model uncertainty related to the HRA to include consideration of the accuracy and completeness issues noted in NUREG-6850, Volume 2, Appendix V.	This F&O has not been resolved. The FPRA includes various actions that are being included as required plant changes in the NFPA 805 LAR. The development and implementation of related procedures has not yet been initiated as it is part of the overall integrated process associated with transition to an NFPA 805 license basis.
7-8	Dependency between multiple altered events representing new HFEs in the same cutset and between action represented by the altered events and other HFEs in the same cutset has not been assessed based on the assumption that the dependency effects are bounded by the application of conservative screening values. However, there is no documented assessment to support this assumption. There are cases where complete dependency between events may be appropriate. For example, cutsets 40 – 45 in the provided Aggregate CDF_aggregate.cut file contain altered events MAVC4200A_1.00E-01 and MAVC4460_1.00E-02 in each cutset. The product of these two events is therefore 1.00E-03. However, since both events involve failure to isolate the letdown line, it could be assumed that there is complete dependence between the events since they would share a common cue. This F&O supersedes 2010 Peer Review F&O 6-16.	HR-H3 HR-I2 HRA-D2 PRM-B11	The dependency associated with operator actions applied using the altered events method has not been addressed. Address dependency between multiple altered events representing new HFEs and between the altered events and other HEPs in the same cutset. If detailed dependency analysis is not performed, provide a justification supporting the assumption that the values chosen for the altered events bounds dependency effects.	This F&O has been resolved. The use of altered events as a surrogate for a recovery action has been significantly reduced as noted previously. Those remaining instances are addressed by modifications to the recovery rule file so that only a single instance of this use would exist in any cutset. This eliminates the potential for multiple surrogate recovery events to appear together in the same cutset.

	(This F&O originated from SR HR-H3)			
8-3	Attachment U – Internal Events PRA Quality (DRAFT), document applicability of Internal Events F&Os to internal events PRA, but not to Fire PRA. There was no evidence that the review of F&O disposition status addressed the question of whether the disposition that was taken would adversely affect the development of the fire PRA. This F&O is derived from 2010 Fire PRA peer review F&O 4-4. (This F&O originated from SR PRM-B2)	PRM-B2	The potential effect of internal events F&O disposition on development of the FIRE PRA was not addressed. Review internal events F&Os and provide documentation as to how disposition of those F&Os may impact development of the Fire PRA.	This F&O has been resolved. The internal events PRA model F&Os that have not been resolved/closed have been reviewed and found to have no negative impact on Fire PRA results or this application.
8-5	The Fire PRA model changes were constructed so as to allow credit for the current internal events PRA model structure using existing accident sequence progression, success criteria and timing. The internal events HRAs are modified with a screening modifier. Travel paths are considered in the human failure evaluation report. However, there is no indication that a review was performed to identify accident sequences that may require modification based on unique aspects of the plant fire response procedures. For example, RWST drain down may affect the evaluation of timing for aligning sump recirculation, which is not presently represented in the non-LOCA transient event tree accident sequences used for the majority of the fire scenarios. A review should be performed for possible changes to success criteria, particularly due to model changes from the MSO evaluation. This F&O is derived from 2010 Fire peer review F&O 6.3. (This F&O originated from SR PRM-B5)	HRA-B1 HRA-B3 PRM-B5 PRM-B7	The standard requires a review of the fire-induced initiating events accident sequences, and success criteria included in the internal events model, to identify new accident sequence progressions or success criteria due to unique aspects of fires. This review will help assure that there are no revised actions where the screening multipliers are not appropriate. Conduct and document a review of the fire-induced initiating events accident sequences, and success criteria included in the internal events model, to identify new accident sequence progressions or success criteria due to unique aspects of fires.	This F&O has been resolved. The review of results and cutsets that were performed did identify a number of instances such as that specifically identified in the F&O. In all instances, it was determined that the existing model structure was appropriate and that opportunities for recovery actions were limited either because of a lack of appropriate cues or insufficient timing to gain any meaningful benefit via recovery. The analysis documentation of the HFE treatment was updated to address the internal events PRA model human actions that are used in the FPRA. The documentation addresses the applicability, numerical adjustment, and availability of necessary cues.
8-8	Several portions of the analysis are not documented for Unit 3. Specific examples include: Unit 4 significant contributors are identified in 0493060006.005, Rev. 2. Unit 3 significant contributors are available, but not fully documented. Unit 4 Fire Scenario information is presented in 0493060006.004, Rev. 2, but the equivalent Unit 3 information (Attachment D and E) is not provided. (This F&O originated from SR PRM-A3)	FSS-E3 PRM-A3 PRM-C1	Unit 3 results not fully documented, although they are available for inspection using the quantification software. Document Unit 3 results consistent with the Unit 4 results.	This F&O has been resolved. The U3 results have been added to the analysis documentation.
8-10	2010 FPRA peer review F&O 1-44 finds issues with masking fire effects by setting basic events to 0 in the Altered Events table. This has partially been addressed by setting these events to 'nominal' in the Altered Events table and for reviewing cases where the nominal value is on one side of an AND gate and the modified HEP value is on the other side. However, there are still cases where fire impacts are masked when the nominally adjusted event is on both sides of an AND gate or the HEP event is on one side of an	HRA-C1 PRM-B11	This appears to have a significant non-conservative impact to PRA results. Given the actions in the altered events report are being added to the model as needed recoveries in order to ensure risk is low, and given the resulting recovery actions do not show up in the results in most cases, there appears to be a disconnect between the addition of new actions to the procedures and the quantification of these actions in the FPRA. It appears part of the disconnect is that the logic modeling, as modified by the altered events table, results in	This F&O has been resolved. The use of '0'; has been eliminated in the Altered Events table. Instead, events are set to nominal. In the case of the application calculations for NFPA 805, the 'compliant' case is determined by using a '0' value which would under-estimate the compliance case risk and thereby provide a conservative estimate of the risk increase for the

	AND gate and nominally adjusted events are on the other side. Scenarios 030 PTB and 067E PTB are two examples. This old F&O is converted to a new F&O 8-10. (This F&O originated from SR HRA-C1)		the recovery values being screened from the results. Due to the complexity of this methodology, it seems a difficult task to review and address for these masking issues. Perhaps a more systematic and comprehensive approach, with an independent review, could provide confidence that these nonconservatisms are addressed. Adding new HEP basic events, consistent with the approach used for internal events, would address this issue.	application.
9-1	A general screening based on the ability to form a damaging HGL in an exposing compartment was developed. If no damaging HGL could form in an exposing compartment then there was no possible associated MC scenario. When a damaging HGL could form, a second screening was performed whereby the frequency of developing the HGL was determined. If the frequency was less than 1E-07/yr, then the scenario could be eliminated. However, there was no basis provided for the 1E-07/yr criteria, nor was the criteria adhered to; in fact, the criteria was exceeded, yet still applied, in over 150 different scenarios. Some of the screened scenarios were slightly over the 1E-07/yr threshold, while others ranged as high as nearly 6E-07/yr. The impact of exceeding the criteria cannot be determined as no specific MCA scenarios were ever developed; therefore, it is not known if the scenarios would be significant. For example, if it is assumed that 10 of the scenarios with a frequency of 5E-07 had CCDPs of 1.0, this would result in an increase in total CDF of 5E-06 which is about 10% of the total fire CDF. (This F&O originated from SR FSS-G2)	FSS-G2 FSS-G3 FSS-G6	No basis for the screening criteria is described. Since no MCA scenarios are developed, there is no way to determine if the exceeded (yet applied) screening criteria are significant. Provide a basis for the 1E-07/yr screening criteria including additional information required when the criteria are included. The basis for the screening criteria should ensure the frequency is not too high, thereby potentially masking significant MCA scenarios. Evaluate impact of exceeded screening criteria; for example, a qualitative analysis of the expected CCDP based on known targets in the exposing and exposed compartments.	This F&O has been resolved. The existing HGL/MCA analysis includes a number of occurrences where the simplified screening approach was found to generate over-conservative results. Incrementally enhanced treatments were applied to confirm that these locations had a very low likelihood of creating or causing formation of HGL conditions and consequently a possible multi-compartment scenario
9-4	The multi-compartment analysis assumes a bounding value of 7.4E-3 for evaluation of active fire barrier elements. Actual fire barrier elements are not considered; instead the failure probability of a fire door is assumed for active barrier element failure because this failure probability represents the highest single probability of a single barrier failure. This method ignores the potential for multiple fire barrier elements. Per NUREG/CR-4840 (source document for NUREG/CR-6850 Table 11-3, "Barrier Types and Their Failure Probabilities") the total barrier failure rate is a union of the probabilities of the individual failure rates. Therefore, a value of 7.4E-03 may be conservative or non-conservative. This is based on 2010 FPRA peer review F&O 1-35. (This F&O originated from SR FSS-G5)	FSS-G4 FSS-G5	Systematic generic assessment of active fire barrier elements may lead to non-conservative results. If a screening value is desired, NUREG/CR-6850 Section 11.5.4.4 suggests using a screening value of 0.1 for active fire barrier elements. This value is much more likely to encompass multiple fire barrier elements. For scenarios that do not screen out, actual fire barrier elements identified during walkdowns (or document review) can be used to develop a more realistic barrier failure probability.	This F&O has been resolved. The update of the analysis to incorporate a barrier failure probability that integrates all possible barrier elements was found to result in a value of approximately double the current value. However, since the entire analysis approach involves a screening strategy, additional analysis refinements are possible. An assessment of the use of a higher barrier failure probability to account for failure of multiple barrier elements found that the overall conclusion that MCA scenarios are not risk significant and need not be explicitly included in the FPRA was confirmed. However, the analysis documentation has not yet been updated to reflect these results and insights.
9-5	The screening criteria are defined in the Turkey Point Hot Gas Layer and Multi-Compartment Analysis, (Report	FSS-G2	Use of a standard fire scenario may be non-conservative for some zones.	This F&O has been resolved. The use of the 5 minute delay to combustible cable

	H0493060006.006) methodology. Compartments that don't screen are retained for further analysis. A concern identified with the screening criteria involves the use of a standard fire scenario for each analysis rather than determining the most challenging fire scenario inherent to the analyzed compartment. This approach potentially masks the potential for forming an HGL in the exposing compartment. For example, in zones 67 and 68 the standard fire scenario is non-conservative due to the potential for HEAF in 4kV switchgear. The damage time of 5 minutes is non-conservative for HEAF scenarios (should use 0 minutes). This F&O supersedes 2010 FPRA peer review F&O 3-11. (This F&O originated from SR FSS-G2)		Review zones to ensure that the standard fire scenario is actually the most challenging scenario inherent to the analyzed compartment. For zones where the standard fire scenario is not the most challenging, determine the most challenging scenario and evaluate accordingly.	ignition is considered realistic. Other conservatisms in the analysis ensure the overall conservatism of the MCA/HGL evaluation.
9-6	The system unavailability records for the plant have not been reviewed in crediting fire detection and suppression systems. This F&O supersedes 2010 FPRA peer review F&O 2-26 (This F&O originated from SR FSS-D7)	FSS-D7	This is a systematic issue. The intent for Capability Category II is to additionally require a review of plant records to determine if the generic unavailability credit is consistent with actual system unavailability. Outlier experience would be any experience indicating that actual system is unavailable more frequently than would be indicated by the generic values. Consider performing and documenting the review of plant records to determine if the generic unavailability credit is consistent with actual system unavailability. Outlier experience would be any experience indicating that actual system is unavailable more frequently than would be indicated by the generic values.	This F&O has been resolved. The fire protection system availability data for PTN has been reviewed and no outlier behavior has been identified.
9-10	Section 3.1 of the FSS Report (0493060006.004, Rev. 2) states: "For the electrical panel fires, the scenarios are developed similar to scenarios involving electrical panel fires outside the Control Room and are adequately described in Attachment A. Fire spread to adjacent panels was determined for these scenarios based on a walkdown of the control room during which panels with potential barriers for spread of fire were opened to confirm the existence of such barriers. For MCB fires, the method from NUREG/CR-6850 Appendix L is applied. NUREG/CR-6850 Appendix L defines a non-suppression probability applicable to the MCB. From Figure L-1 of NUREG/CR-6850, for non-qualified cables, and for a bounding distance of 0 meters (assuming that the cables terminating at the individual MCB are in very close proximity), a non-suppression frequency of 8.30E-3 is used for the MCB." However, based on discussion with FPL/ERIN staff, this was not done. Essentially, no fire spread for any cabinet in the MCR was assumed. For panels with incipient detection, success of the detection results in no damage as it is assumed operators isolate the circuit prior to additional damage in the cabinet. If insipient detection fails, the MCB panel fails completely, but never spreads another cabinet. For all other cabinets/MCB panels without incipient	FSS-A6 FSS-H7	Assumption made that no cabinet/panel fires in the MCR will ever spread to an adjacent cabinet even if the cabinets are open to one another. This incorporates an implied assumption that every MCR panel/cabinet fire will be extinguished prior to spread. Identify adjacent MCR cabinets/panels which could result in fire spread given failure of suppression. Apply NUREG/CR-6850 Appendix L, S or other relevant document to address the potential for fire spread. Ensure documentation is consistent with the process actually used in the analysis.	This F&O has been resolved. Panels with communication between adjacent panels are to be provided with incipient detection to ensure early identification of fire to preclude spread between panels.

	detection, full burnout of the cabinet/panel is assumed, but again, no spread to adjacent cabinets is assumed even if the cabinets are open to one another (e.g., walkthrough MCB). (This F&O originated from SR FSS-A6)			
9-11	Several entries in Table 3-1 of Report H0493060006.006 says "Walkdown required to confirm no combustibles within the 383 ZOI" when the cables are not IEEE-383 qualified. Based on discussions with FPL and contractors, it is believed that this is a typo, and the correct damage criteria were actually applied. (This F&O originated from SR FSS-G2)	FSS-G2	It was not confirmed during the peer review that this was just a documentation issue. Therefore, this is classified as a finding because it could affect the analysis results. Verify that the damage criteria used is consistent with non-383 cable damage and revise the documentation as required. If it is discovered that the incorrect damage criteria were applied, update the analysis with the correct criteria.	This F&O has been resolved. It was confirmed that the lower damage threshold associated with thermoplastic materials was used for the analysis. The typographical error has been corrected.
10-1	The 2010 peer review identified that "Fire modeling was conducted via generic fire modeling from which Zones-Of-Influence (ZOI) for specific initiator types was generated. The ZOIs were used to define bounding fire characteristics for each fire scenario. Characteristics that are used to bound potentially risk contributing fire events are identified in Attachment B of the Fire Scenario Report, (Report 0493060006.004). Based on the use of a bounding approach this SR is judged to be met at CC I. Significant fire scenarios should be developed with 2-point fire modeling." Since this review, FP&L has stated that "The use of a panel split fraction to differentiate between fires impacting the panel and components with cables terminating at the panel versus panel fires impacting cables outside of the panel provides an equivalent and more useful two point fire model." The Panel Split fraction is developed from a supplemental report (ERIN report, Supplemental Fire PRA Methods, dated February 2010). This document was submitted to the EPRI Fire PRA Methods Review Panel. This review is not complete as of the date of this peer review. Use of the split fraction method is based on industry events rather than site specific fire ignition sources and target configurations and therefore, could result in non-conservative frequency estimates of target damage. (This F&O originated from SR FSS-C1)	FSS-C1 FSS-G1	The present analysis provides a bounding approach for fire severity in most cases, since the 98th percentile fire heat release rate is used. However, use of the split fraction method is based on industry events rather than site specific fire ignition sources and target configurations. Therefore, this could result in non-conservative frequency estimates of target damage. Perform 2-point fire modeling, when applicable, for risk significant fire scenarios.	This F&O has been resolved. The recommended resolution action in the F&O was assessed in the context of the dominant fire risk contributors. This assessment concluded that further refinements such as that described in the F&O would not substantively change the results of the analysis. The existing treatment retains some conservatism which results in this SR meeting CC I. This is adequate for the 4b application, as this conservative bias would tend to over-estimate the risk. The issue regarding the ERIN panel split fraction is addressed in the disposition for F&O 10-3.
10-2	The 2010 review of PTN Tasks 8 and 11 Report 0493060006.004, identified that 'no hydrogen fires other than turbine/generator have been postulated.' (Previously F&O 5-16) Since this Finding was identified, FP&L has determined that 'Miscellaneous Hydrogen piping at PTN is limited to hydrogen supply to the VCT tanks. The associated piping is located in the charging pump rooms (Fire Zones 45 and 55). Fires in these fire zones are assumed to impact all components in the fire zone. The associated risk is low	FSS-A1	Including the fire frequency and associated fire scenarios from hydrogen fires will have impact to the CDF and LERF results. Incorporate the hydrogen fire scenarios being developed into the model, and update documentation as necessary.	This F&O has been resolved. Miscellaneous hydrogen fires have been incorporated in the Fire PRA in the charging pump room fire areas where the hydrogen lines associated with VCT cover gas are routed.

	given the availability of thermal barrier cooling for RCP seals and HHSI pumps. Allocation of the IGF associated with miscellaneous hydrogen fires to these fire zones would result in an increase in the ignition frequency for these zones by less than a factor of 3. Given the low risk significance of these zones this will have a negligible impact on overall plant risk and the charging pump rooms will remain low risk contribution fire zones. Incorporation of this ignition frequency into the associated documentation will be incorporated in a future revision to the documentation.' Hydrogen fires are also being developed for H2 piping and valves in Compartments 82 and 87 (scenarios 82-P and 87-P). However, since these do not appear yet in the Fire Scenario Report, action is required. This finding is currently being addressed and appears to be resolved once the new H2 fires are included in the model and documentation is updated. (This F&O originated from SR FSS-A1)			
10-3	FSS-C4 requires severity factors to be independent of other factors. Fire severity factor as discussed in Section 7.1.2 for electrical cabinets is not developed or applied consistently with the NUREG/CR-6850 methods. This is developed from a supplemental report (ERIN report, Supplemental Fire PRA Methods, dated February 2010). This document was submitted to the EPRI Fire PRA Methods Review Panel. This review is not complete as of the date of this peer review. Using this method, fire propagation outside of the electrical cabinets is dependent on the nonsuppression probability. Therefore, some dependency exists in this data if used in conjunction with a non-suppression factor. Due to this derivation of the conditional probabilities for fire propagation outside of the cabinets, the conditional probabilities thus developed (and applied in the FRANC model) could potentially be non-conservative. The severity factors are developed using generic fire events data from the EPRI fire events database. Given the fire data duration and damage is a result of multiple factors (growth, suppression, severity, location, etc.), and given the fire data often does not have sufficient information to make a reasonable determination of either the fire size or whether a fire propagated outside the cabinet, the severity factor used (panel split fraction) may not necessarily bound the conditions of the specific fire scenarios under analysis. (This F&O originated from SR FSS-C4)	FSS-C4 FSS-D5 FSS-G1	Severity factor (panel split fraction) is used extensively in the Fire PRA. Use the severity factor method described in NUREG/CR-6850, or develop an accepted industry approach (presently being discussed by EPRI). Develop fire severity factors based on the likely HRR and location of overhead cables or location of equipment. For example, if cable is 7 feet overhead, the severity factor would be based on the minimum HRR that would damage the cable at that distance. Additionally, the growth time can be used in determining non-suppression time.	The FPRA quantification uses the panel factors consistent with the latest guidance from the EPRI Methods Review panel. A sensitivity study has been performed to address the impact of elimination of the credit for the panel factors. The results of this evaluation indicate that the delta CDF/LERF would exceed the Reg Guide 1.174 guidelines should these factors be completely eliminated (the 1E-5/1E-6 delta CDF/delta LERF limits would be exceeded but the conservatively calculated delta risk would be less than 2E-5/2E-6). Further refinements of this sensitivity evaluation are possible to reduce the calculated delta risk. Credit for additional Defense In Depth measures may be taken in areas of concern as necessary to compensate for the increased delta risk.
10-4	One situation was identified for which credit of fire wrap is taken in Compartment 96 for ignition source 3B04, which is a 480V load center. This fire wrap protects PB3319, PB3813, PB7022, and PB7521. The wrap appears as being	FSS-C8	This finding is based on identification of credit for a wrap in Attachment A of the Fire Scenario Report, (Report 0493060006.004). Any credited fire wrap should be addressed and the wrap integrity should be established with respect to	This F&O has been resolved. A qualitative assessment has been performed to assess the potential impact of this F&O. The hose stream test imposed on the fire barrier

	credited in a HEAF scenario. No justification for crediting this wrap assuming mechanical damage and direct flame impingement from the HEAF is provided. Similar issue for 3B03 also in Compartment 96. Thermo-lag is also seen as credited in some scenarios, which would require justification due to issues with this particular type of cable barrier. (This F&O originated from SR FSS-C8)		fire resistance, mechanical protection, and potential fire related exposure to which the wrap may be exposed (direct flame impingement, HEAF, etc.).	qualification subsequent to fire exposure is considered to provide a comparable level challenge to the thermolag barrier as would the HEAF force applied at the onset of fire exposure.
10-6	Treatment for transient fire damage to targets is measured from the compartment floor rather than the height of the transient fuel package that is typically considered. Discussion with FP&L during the review provided some basis for the damage height (indicating that transient fires above the floor will have an overall lower average surface HRR). However, the supplemental discussion was still considered inconsistent with past events and existing guidance on analysis of transient fires, and could lead to non-conservative estimates of transient fire damage to targets. Transient fire evaluations conducted as described in the Fire Scenario Report result in screening fire damage to targets that are located > 7.3' above the floor which is believed to be non-conservative for developed fires involving ordinary combustible fuel packages such as a trash can or trash bag. In response to this concern it was pointed out that the thermal plume component relies on empirical relationships between the source strength and the distance between the virtual origin of the fire and the target. The fire plume begins to entrain air at the lowest point of burning, which defines the base of the fire; normally at the floor. However this argument ignores the potential that a fire could begin burning at the top of a fuel package thus elevating its base. At a minimum, during the initial period of burning, damage temperatures generated by the fire would likewise be elevated. Over time the base of the fire may change due to collapse of the fuel package or burning away of the fuel, however the empirical model presented did not present sufficient basis for assuming that the base of the fire is at the floor for its entire duration. (This F&O originated from SR FSS-D6)	FSS-D6	Many transient fire scenarios have been screened during detailed scenario analysis. The results of the FPRA are therefore potentially non-conservative for the analyzed detailed scenarios. The transient fires should be considered to be above the floor level in the analysis.	This F&O has been resolved. Supplemental walkdowns were performed to re-assess the treatment of transient fires. These walkdowns focused on two key attributes – the appropriateness of the selected HRR characterization and the location of the postulated fire scenarios. With respect to this specific F&O, the placement (elevation) of the assumed fire was based on the physical features of the location. The fires were not artificially elevated in the absence of a physical feature.
10-8	Ambient conditions are assumed in the Generic Fire Modeling Treatment Report (prepared by Hughes). Ambient temperature is assumed to be 68°F for all calculations. No technical discussion or justification is provided in the Fire Scenario Report to substantiate that this is a reasonable value for the compartments where this was applied. (This F&O originated from SR FSS-D4)	FSS-D4 FSS-H4	Underestimating the ambient conditions could result in non-conservative estimations of zones of influence and targets considered to be fire damaged. Assess areas where elevated ambient temperatures could be experienced and justify the acceptability of the models used. Otherwise, incorporate elevated ambient temperatures into the zone of influence calculations.	This F&O has been resolved. A qualitative assessment has been performed to assess the potential impact of this F&O. The sensitivity of the ZOI dimensions to the ambient temperature is relatively low as described in the original Hughes Generic Fire Modeling treatments report, in particular for IEEE-383 qualified/Thermoset cables. In the case of an initial ambient temperature of 35°C, the expected affect on

				the ZOI dimensions is within the measurement uncertainty in the field.
10-9	The 2010 peer review identified that 'Except for the MCR fire scenarios, no other fire scenario has used the Non-Suppression Probability (NSP) in PTN fire model at this time.' Since this review, FP&L has taken credit for suppression (both automatic and manual) in the Multi-Compartment/Hot Gas Layer evaluation. However, this evaluation does not include an assessment of the fire protection system effectiveness. Of particular concern is that fire detection and/or suppression timing (i.e., thermal response of the detector and/or sprinkler) was not calculated and subtracted from the time considered for manual suppression when using the FAQ-0050 process. In addition, fire detection reliabilities are not included in the assessment. If the detection system does not function as intended, the time to detection to initiate fire brigade response would be substantially longer. (This F&O originated from SR FSS-D8)	FSS-D8	The method currently employed could result in optimistic times for suppression activation or fire brigade response. Assess and document the effectiveness of suppression with respect to: System design complies with applicable codes and standards, and current fire protection engineering practice, The time available to suppress the fire prior to target damage, Specific features of physical analysis unit and fire scenario under analysis (e.g., pocketing effects, blockages that might impact plume behaviors or the "visibility" of the fire to detection and suppression systems, and suppression system coverage), and Suitability of the installed system given the nature of the fire source being analyzed.	A qualitative assessment has been performed to assess the potential impact of this F&O. The HGL and MCA analyses credit both automatic suppression system and fire brigade actions. In the context of the HGL and MCA, the fire brigade action of interest is fire control as that would terminate the possibility for HGL formation. However, the only readily available numeric credit is fire suppression credit. To reduce the conservatism introduced into the analysis, fire detection time is ignored for the HGL and MCA. The timeframe associated with detection and suppression is significantly less than the timeframe required to reach a hot gas layer temperature which would impact the HGL analysis.
10-11	The 2010 peer review identified that "fire scenario evaluation tools were developed based on the Generic Fire Modeling Treatments. These walkdown/evaluation tools are based on bounding fires that are assumed to cause target damage at a height above the base fire with the fire burning at peak intensity and without burnout times. Because these tools assume a fire burning at peak intensity and without burnout, this SR is considered met at CC I." Since the review, FP&L has stated that "The use of a panel split fraction to differentiate between fires impacting the panel and components with cables terminating at the panel versus panel fires impacting cables outside of the panel provides an equivalent and more useful two point fire model... The application of the two point treatment to individual fire scenarios is carried through to the MCA/HGL evaluation which addresses the impact of each scenario on MCA." The Panel Split fraction is developed from a supplemental report (ERIN report, Supplemental Fire PRA Methods, dated February 2010). This document was submitted to the EPRI Fire PRA Methods Review Panel. This review is not complete as of the date of this peer review. Use of the split fraction method is based on industry events rather than site specific fire ignition sources and target configurations and therefore, could result in non-conservative frequency estimates of target damage. (This F&O originated from SR FSS-C2)	FSS-C2 FSS-C3 FSS-G1	The present analysis provides a bounding approach in most cases, since the 98th percentile fire heat release rate is used from fire initiation without growth and burnout. However, use of the split fraction method is based on industry events rather than site specific fire ignition sources and target configurations. Therefore, this could result in non-conservative frequency estimates of target damage. Include fire growth and decay for risk significant fire scenarios.	This F&O has been resolved. The recommended resolution involves the crediting of growth and decay in the modeling of the postulated fire. The existing analysis does not take credit for these variables. A review of the dominant fire scenarios found that the risk benefit that might be gained is minimal. Therefore, this refinement was not performed. The resulting categorization of the related SR is CC 1. Since the approach results in some conservatism being retained in the results, this CC is judged to be adequate for the NFPA 805 applications as the conservative bias would tend to result in the over-estimation of the risk metrics used for this application.
10-12	The 2010 peer review identified that "The PTN FPRA methodology generally does not include postulation or	FSS-D9	This appears to be a documentation issue, but FP&L should confirm that smoke damage has been considered and	This F&O has been resolved. An analysis of the impact of smoke damage has

	evaluation of smoke damage. Additional review shows that the smoke issues do not affect the FPRA results significantly. However, the FPRA does not include a qualitative evaluation of smoke damage to FPRA equipment." Since the 2010 review, FP&L stated, "Section 6.2 of the Scenario Report was added to address this concern." However, section 6.2 provides a high level discussion and methodology including the statement that "Exposure time plays a key role in the likelihood of failures from smoke. As a result, damage from short term smoke exposure will only result from severe conditions.... Instruments, control components and all high voltage powered components are exceptionally vulnerable to circuit bridging as a result of airborne smoke and deposited particulates." However, there is no documented discussion of the smoke damage assessment results, and none of the targets in the scenarios indicated smoke damage as the failure mode. NUREG/CR-6850 recommends considering smoke damage to banks of interconnected panels, and this should be considered. (This F&O originated from SR FSS-D9)		document accordingly. Confirm that smoke damage has been considered and document accordingly.	been completed and documented in the PTN FPRA Scenario Report.
10-13	A credit for incipient detection is taken for MCB fires (non-suppression probability of 0.02). There is no documentation to justify this value. Per discussion with FP&L the approach appears to be in agreement with FAQ-08-0046. The approach also does not use the NUREG/CR-6850 Appendix L factor for panels that credit incipient detection. Secondly, the incipient detection system is not yet installed, and therefore, the Fire PRA should be reviewed and updated as needed to reflect any differences between the assumed and as-built conditions of the system. (This F&O originated from SR FSS-A6)	FSS-A6 FSS-D7 FSS-H7	This appears to be a documentation issue, but since the system is not yet installed, there could be an impact to the assumptions made within the Fire PRA. Document the basis for probability of non-suppression value assumed in analysis. When the incipient system is installed, the FPRA should be reviewed and updated accordingly.	This F&O has been resolved. The credit taken for incipient detection is consistent with that specified in FAQ-08-0046.
10-14	Beyond the Generic Fire Modeling Treatments, the Fire PRA did not include additional detailed fire modeling for most fire compartments. Note 4 (under FSS-A5 of the ASME Standard) states that "once a fire scenario has been 'selected,' this implies that the scenario will eventually be evaluated and/or quantified at a level of detail commensurate with the risk significance of the scenario." (This F&O originated from SR FSS-A5)	FSS-A5	For risk significant fire scenarios, detailed fire modeling should be performed to ensure you are not masking the "true risk significant fire areas". Without detailed fire modeling for significant fire scenarios, the results are conservative. Consider performing additional detailed fire modeling to provide "reasonable assurance that the fire risk contribution of each unscreened physical analysis unit can be characterized."	This F&O has been resolved. The current analysis is consistent with a Capability Category I analysis. This provides a degree of conservatism in the analysis which would also tend to over-estimate the change in risk which is reported for the NFPA 805 application. A review of the results of the application analyses indicates more rigorous analyses consistent with CC II or CC III would not alter the conclusions of the analyses.
10-15	PTN credits multiple suppression paths for MCA/HGL evaluation. However, the dependencies have not been evaluated and modeled. For example, fixed suppression and fire brigade response may both rely on a single detection system.	FSS-C7 FSS-G1 FSS-H7	Lack of dependency analysis could lead to an optimistic estimate of suppression probability. When multiple suppression paths are credited, perform a review and address any dependencies between suppression and detection systems credited in the MCA/HGL calculation.	This F&O is resolved. A review of the credited suppression systems in the Multi-Compartment /Hot Gas Layer analysis has confirmed that no dependency exists between the suppression systems and detection systems. Detection in the zones with suppression systems is

	(This F&O originated from SR FSS-C7)			associated with an independent detection system.
10-16	Review of fire modeling in single compartments does not consider the addition of HRR from secondary combustibles. It is acknowledged that secondary combustibles were considered for the MCA/HGL evaluation. Fire spread and additional HRR due to the resulting cable tray fire and adjacent cabinets would increase the total fire size and the subsequent zone of influence. Compared to the NUREG/CR-6850 guidance for flame spread along PVC cable (flame spread = 0.9 mm/sec) the estimation of HRR for the applied scenarios is non-conservative. Realistic estimation of the scenario HRR is necessary to ensure the full impact of the fire on exposed targets is presented and that the effects of a damaging HGL may also be estimated. FP&L has stated that walkdowns are in progress to include fire spread to cable trays and incorporate this into the fire scenarios. (This F&O originated from SR FSS-C1)	FSS-C1 FSS-D3 FSS-G1	Discounting of secondary combustibles when considering localized fire damage could lead to non-conservative results. Include secondary combustibles in the heat release rates used for zone of influence estimates.	This F&O has been resolved. Supplemental walkdowns have been performed to identify and address the potential for fire spread for scenarios where the non-383 cables are not protected by Flammastic material. The analysis has been updated to include these scenarios as appropriate.
10-17	The 2010 peer review identified that Attachment B of the Fire Scenario Report (Report 0493060006.004) generic fire modeling treatments do not account for the effects of hot gas layer (HGL) on the zones of influence. The limitation indicates that because HGL is not considered that these correlations should not be used in enclosed areas with small volumes where a significant HGL thickness may form. Because this relationship is not considered plume temperatures may be underestimated because it is assumed that ambient temperature air is being entrained into the plume, resulting in cooler plume temperatures, rather than heated air from the hot gas layer. Entrainment of heated air into the fire plume results in higher damage heights because the plume remains hotter at higher elevations. Since this review, FP&L states that "The impact of a hot gas layer on the zone of influence is evaluated for all fire zones/scenarios in the MCA/HGL evaluation." A review of this evaluation confirms that HGL effects on ZOI were in fact considered for the generic treatments; however, there is not sufficient documentation in Attachment A to the Fire Scenario Report to determine which zone of influence was applied to which scenario, and whether it was applied correctly to consider the effects of HGL. The Generic treatments include several iterations and combinations of variables, including opening percentage of the compartment. The fire scenario documentation at the time of this review did not provide sufficient information on opening percentage to confirm that the ZOI was applicable to the compartment.	FSS-D1		This F&O has been resolved. The MCA/HGL evaluation has been modified to address the concern noted in the F&O. The potential for a larger zone of influence is addressed via new fire scenarios added to the fire PRA.

	(This F&O originated from SR FSS-D1)			
10-18	In at least two cases, transient fire scenarios have not been included in the fire modeling for some compartments (e.g., fire compartments 67 and 68). Per discussion with FP&L the transients may have been excluded based on the dominance of the frequency of fixed scenarios. However, transients should only be excluded when precluded by design. Based on the size of these rooms, and the presence of secondary combustibles, transient fires could lead to fire growth and eventually HGL, and therefore should be analyzed. (This F&O originated from SR FSS-A1)	FSS-A1	The exclusion of transients in some compartments may lead to a non-conservative estimate of CDF and LERF. Include transient scenarios in all compartments where fire modeling has been employed.	This F&O has been resolved. Supplemental walkdowns were performed to re-assess the treatment of transient fires. These walkdowns focused on two key attributes – the appropriateness of the selected HRR characterization and the location of the postulated fire scenarios. The postulated location for the treatment of transient fires was based on where a transient ignition source might reasonably occur. The results of these walkdowns were incorporated into the FPRA analysis.
10-19	For fire modeling analysis of transient fires, FP&L implements a floor area weighting factor. However, the documentation does not include a graphical representation of the assumed transient locations and boundaries. It is therefore not possible to review (or update) transient fires. Also during review of transient weighting factors it appears to have been double counted in some compartments (e.g., compartment 63). Based on discussion with FP&L this was due to an error in the Excel based spreadsheet tool for transient frequency quantification. This appears to be an isolated case and will be corrected. (This F&O originated from SR FSS-H1)	FSS-H1	Lack of documentation on transient fire locations and boundaries will present a challenge for updates and peer reviews. Update documentation to include a graphical representation of transient fire locations and boundaries.	This F&O has been resolved. The specific instance noted in the F&O was corrected. In addition, supplemental walkdowns were performed to re-assess the overall treatment of transient fires. These walkdowns focused on two key attributes – the appropriateness of the selected HRR characterization and the location of the postulated fire scenarios. However, the documentation that was generated did not specifically produce graphical representations. Instead, the information was incrementally enhanced to provide a spatial reference to a location with in the space. The need for special depiction of transient fire scenario locations will be addressed in conjunction with the development of procedures for post transition configuration control.
10-20	The fire modeling analysis of the Turbine Generator (T/G) fires is performed in accordance with Appendix O to NUREG/CR-6850. However, there is no discussion regarding the lack of analysis of the catastrophic T/G fire event, which should consider blade ejection, oil line rupture, and hydrogen explosion. Per discussion with FP&L, the catastrophic fire was discounted since the T/G is located outdoors. While this may not result in hot gas layer formation and structural collapse, a review of the guidance is warranted, and inclusion of this event frequency should as a minimum map to the loss of the T/G and if suppression fails, all equipment within the T/G structure. (This F&O originated from SR FSS-A1)	FSS-A1	Lack of consideration of the catastrophic T/G fire may lead to a non-conservative estimate of CDF and LERF. Perform a review of the catastrophic T/G fire in accordance with Appendix O to NUREG/CR-6850, or document the justification for excluding this event at PTN.	This F&O has been resolved. The analysis documentation has been updated to address catastrophic T/G fires that may lead to building collapse or other significant widespread damage. The results of this update did not identify any new risk significant contributors or insights.
10-21	The supplemental generic Fire Model Treatments: Transient Ignition Source Strength includes an assumption for transient burnout of 12 minutes. This burnout time is based on an assumed fire loading and the 317kW heat release rate, and appears to be optimistic given the uncertainty in transient fire loading. The burnout is then used to develop a	FSS-C3 FSS-G1 FSS-H2	The current approach results in many transient fire scenarios being screened during detailed scenario analysis. The results of the FPRA are therefore potentially non-conservative for the analyzed detailed scenarios. Provide additional justification for the applied transient fire analysis as a screening approach. Consider increasing the burnout time and using the	This F&O has been resolved. Supplemental walkdowns were performed to re-assess the treatment of transient fires. These walkdowns did not identify any instances where an altering of the transient fire duration had any material impact on the HGL and MCA. The

	zone of influence for thermoplastic targets, based on the thermal response tables in Appendix H to NUREG/CR-6850 for thermoplastic cable at 260°C. Since this resultant vertical zone of influence is used to screen transient scenarios from impacting secondary targets higher than 7.3 feet from the floor, additional justification is needed to demonstrate that a 12 minute fire, and subsequent use of 260°C damage threshold is appropriate for screening purposes. Also noted is that Attachment B to the Fire Scenario Report zone of influence does not reflect the same values recommended by the Generic Fire Model Treatment. As an example, the differentiation between transient Severe and Non-Severe categories is not based on a 317kW fire. This appears to be a documentation issue only. (This F&O originated from SR FSS-C3)		NUREG/CR-6850 recommended damage threshold to 205°C to bound uncertainties in fuel loading for transient fires.	documentation has also been updated to address the criteria used for selecting the characteristic transient fire HRR. The approach is consistent with the recently issued guidance from the EPRI/NRC review panel. The results of these walkdowns were incorporated into the FPRA analysis. The twelve minute fire corresponds to the 317 kW fuel package only and represents ~ 35 lb. of Class A material. Additional discussion is provided in Rev. 0 of Supplement 3 of the Hughes Generic Fire Modeling treatments that examines the fire durations and test durations of all NUREG/CR 6850 tests. It is shown that the method used to determine a 12 minute fire predicts or overestimates the fire duration in all cases and is therefore a sound approach.
10-22	Per NUREG/CR-6850, appendix H, temperature sensitive equipment should be considered to fail at 65°C. Supplemental Generic Fire Model Treatments: Hot Gas Layer Tables includes new zone of influence and hot gas layer treatments for temperature sensitive equipment. However, per discussion with FP&L these have not been implemented in the fire scenarios. (This F&O originated from SR FSS-C6)	FSS-C6 FSS-G1	For smaller volume rooms, estimates of equipment damage may be non-conservative. Apply the appropriate hot gas layer and zone of influence for temperature sensitive equipment where applicable.	This F&O has been resolved. The consideration of sensitive electronics was addressed in a qualitative fashion in the Scenario Report.
10-23	The PTN FSS report 0493060006.004, Rev 2, section 6 discusses the damage criteria for thermal, smoke, and sensitive equipment. However, suppression effects do not appear to have been considered for the potential to damage equipment. (This F&O originated from SR FSS-C5)	FSS-C5	Equipment damaged by suppression activities may impact estimates of CDF and LERF for some scenarios. Perform an assessment of electrical equipment that may be vulnerable to water intrusion from suppression activities (or thermal shock from gaseous systems), and include any additional failed equipment, not already considered damaged by fire, in scenarios as appropriate.	This F&O has been resolved. The specific issue raised in the F&O is beyond the scope of the associated SR. In addition, no known consensus method exists for treatment. A qualitative assessment, based on other guidance for evaluation of potential impact of suppression effects was performed which indicated that no specific change in the analysis is needed.

References

1. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document*, Nuclear Energy Institute, Revision 0, November 2006.
2. Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2, March 2009.
3. ASME/ANS RA-Sa-2009, *Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008*, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.
4. ML071200238, *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 (TAC No. MD4995)*," Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.
5. Lettis Consultants International (LCI) Project Report Turkey Point Seismic Hazard and Screening Report, .Calculation of Seismic Hazards for CEUS Sites, Project No. 1041, Nicholas Graehl, February 14, 2014.
6. Generic Issue 199 (GI-199) Implications of Updated Probabilistic Seismic Hazard Estimates in Eastern United States on Existing Plants Safety/Risk Assessment, August 2010.
7. Turkey Point Nuclear Plant Probabilistic Safety Analysis Peer Review Report, Final Report, WOG, December 2002.
8. RG 1.200 PRA Focused Peer Review against the ASME PRA Standard Requirements for the Plant Turkey Point Nuclear (PTN) Probabilistic Risk Assessment, Westinghouse Project PA-RMSC-0386R1, July 2011.
9. Letter from Hughes Associates, "Results of focused scope peer review on CCF, ISLOCA, and LERF for Turkey Point Nuclear Generating Station," 11/25/2013.
10. Turkey Point Nuclear Plant Units 3 & 4 Fire PRA Peer Review Report Using ASME PRA Standard Requirements, GE Hitiachi, April 2010.
11. Westinghouse letter LTR-RAM-II-12-037, Follow-on Fire PRA Peer Review Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard for the Turkey Point Nuclear Plant Units 3 and 4 Fire Probabilistic Risk Assessment, May 1, 2012.

Turkey Point Nuclear Plant, Units 3 and 4

Enclosure 3

Information Supporting Technical Adequacy of Probabilistic Risk Assessment (PRA) Models without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 2

This enclosure is not applicable to the Turkey Point Nuclear Plant submittal. Florida Power & Light is not proposing to use any PRA models in its Risk-Informed Completion Time Program for which a PRA standard, endorsed by the NRC in RG 1.200, Revision 2 does not exist.

Turkey Point Nuclear Plant, Units 3 and 4

Enclosure 4

INFORMATION SUPPORTING JUSTIFICATION OF EXCLUDING SOURCES OF RISK NOT ADDRESSED BY THE PRA MODELS

Introduction

Section 4.0, item 5 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" (Reference 2) requires that the License Amendment Request (LAR) provide a justification for excluding any risk sources determined to be insignificant to the calculation of configuration-specific risk, and provide a discussion of any conservative or bounding analyses to be applied to the calculation of risk-informed completion times (RICTs) for sources of risk not addressed by the PRA models.

This enclosure provides information supporting justification of excluding sources of risk not addressed by the Turkey Point PRA.

Scope

NEI 06-09 and the associated Pressurized Water Reactor (PWR) Owners Group (PWROG) guidance (Reference 3) do not provide a specific list of hazards to be considered in a RICT program. However, NUREG-1855 (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:

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

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

RMTS. Consistent with NUREG-1855, the process includes the ability to address external hazards by:

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

The ASME/ANS PRA Standard (Reference 5) has endorsed the following set of five external hazard screening criteria:

(1) The hazard would result in equal or lesser damage than the events for which the plant has been designed. This requires an evaluation of plant design bases to estimate the resistance of plant structures and systems to a particular external hazard.

(2) The hazard has a significantly lower mean frequency of occurrence than another event (taking into account the uncertainties in the estimates of both frequencies), and the hazard could not result in worse consequences than the other event.

(3) The hazard cannot occur close enough to the plant to affect it. Application of this criterion needs to take into account the range of magnitudes of the hazard for the recurrence frequencies of interest.

(4) The hazard is included in the definition of another event.

(5) The hazard is slow in developing, and it can be demonstrated that sufficient time exists to eliminate the source of the threat or to provide an adequate response.

The review of external hazards considers two aspects of the contribution to risk. The first is the contribution from the occurrence of beyond design basis conditions (i.e., winds greater than design). These beyond design basis conditions challenge the functionality of the systems, structures, and components (SSCs) to 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 (i.e., high winds causing loss of offsite power). 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 cause a demand on these systems and can impact configuration risk.

Note that when the effect of a particular hazard is not mitigable using the plant SSCs, then there is no impact on the changes in risk calculated to support the RICT Program, and so these hazards can be screened as well. Only events which create a demand for mitigation equipment are potentially relevant to the RICT Program.

The review and disposition of each external hazard is addressed in Table E4-1. Unless otherwise specified, all information is based on the Individual Plant Examination of External Events (IPEEE) (Reference 6).

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
Aircraft Impacts	There are four airports within the vicinity of the site: Homestead Air Force Base (5 miles), Homestead General Aviation Airport (13 miles), Kendall-Tamiami Executive Airport (15 miles), and Miami International Airport (25 miles). Homestead General Aviation and Miami International were screened from consideration based on the Standard Review Plan (SRP) criteria (Reference 7) methods. The frequency of an aircraft impact at Turkey Point due to air traffic associated with Kendall-Tamiami Executive Airport was calculated as 8.51E-7/year (SRP method), with a more realistic method yielding 1.17E-8/year (Sandia method). Similarly for Homestead AFB, the aircraft impact frequencies were 8.37E-7/year and 2.11E-8/year, for the SRP and Sandia methods, respectively. The most recently available FAA data (at faa.gov) from Airport Master Record shows that annual operations at Homestead General Aviation (2011) and Miami International (2012) were still below the SRP screening criteria. For Kendall-Tamiami Executive Airport, the IPEEE conservatively assumed annual operations would return to pre-hurricane Andrew levels (1992); however, the most recently available FAA data show annual operations	Three of the four nearby airports satisfy the screening criteria of the SRP. The Homestead AFB does not screen out, but the realistic calculation of on site aircraft crash frequency is 1.84E-7/year. If such a postulated event were to occur and cause significant site damage such that it were not able to be mitigated, there would be no impact on configuration risk; otherwise, additional credit for available mitigating systems would further reduce the configuration risk impact. Therefore it is concluded that aircraft impacts do not pose a significant configuration risk, and no unique PRA model for aircraft impact scenarios is required in order to assess configuration risk for the RICT Program.

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
	still well below (194,111 in 2011 vs. 373,500 assumed), and in fact below the SRP screening criteria. Therefore, the estimated aircraft impact frequency for Kendall-Tamiami Executive Airport remains bounding. For Homestead AFB, the most recently available FAA data for annual operations is 49,400 compared to 39,310 assumed for the IPEEE. Updating the aircraft impact frequencies yields 1.05E-6/year (SRP) and 1.84E-7/year (Sandia).	
External Flooding	The external flooding hazard was evaluated as bounded by the storm surge hazard. The plant floodwall provides protection above 18' MSL, which is the minimum flood level required to damage safety related equipment, primarily the 4.16 kV switchgear. The frequency of such an event is estimated to be 1E-6 to 1E-4 per year.	External flooding scenarios do not pose a significant safety impact to Turkey Point based on the design of the facility and external flood protection provided by the plant floodwall. The potential for plant damage due to external flood events is not sensitive to the plant configuration, since events which exceed the 20' above MSL floodwall result in probable loss of all safety-related switchgear. Such extreme flooding events would not be mitigable using plant equipment, and so would not impact configuration risk calculations. It is therefore concluded that no unique PRA model for external flooding scenarios is required in order to assess configuration risk for the RICT Program.

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
Extreme Winds and Tornados (including generated missiles)	The IPEEE included a detailed evaluation of the potential impact of high winds and generated missiles on exposed plant equipment and structures. The possible impacts evaluated included loss of Intake Cooling Water (ICW), unit 3 Refueling Water Storage Tank (RWST), units 3 and 4 diesel generators, Condensate Storage Tank (CST), and instrument air. (Loss of offsite power was assumed to always occur.) The analysis demonstrated that the total CDF was below the 1E-6 per year screening value, and included credit for mitigation by unaffected equipment (Table 5-12 of the IPEEE).	Based on the insignificant impact of these events, it is concluded that no unique PRA model for extreme winds and tornadoes is required in order to assess configuration risk for the RICT Program.
External Fires	Forest fires in the plant vicinity were evaluated as having a minimal potential impact on the plant, and are bounded by the effects of a loss of offsite power.	The impact of an external fire is bounded by the existing loss of offsite power initiating event. It is therefore concluded that no unique PRA model for external fires is required in order to assess configuration risk for the RICT Program.
Accidents From Nearby Facilities	A review of nearby facilities was conducted and identified Homestead AFB and Turkey Point units 1 and 2 (natural gas and oil electric generating plants) as the only nearby facilities requiring evaluation.	Nearby facility accidents do not pose a significant safety impact to Turkey Point based on conformance to the SRP or the very low frequency of the event. It is therefore concluded that no unique PRA model for

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
	Based on a review of satellite images, there are no new facilities in the vicinity of Turkey Point, and so the conclusions of the IPEEE remain valid. The IPEEE evaluated potential toxic chemical releases and explosions due to these offsite facilities, and concluded such accidents either meet SRP criteria and screen out, or represent very low frequency events.	facility accidents is required in order to assess configuration risk for the RICT Program.
Pipeline Accidents (e.g., natural gas)	The only pipeline within 5 miles of Turkey Point is the buried pipeline providing natural gas to Turkey Point units 1 and 2; this pipeline is only above grade at a gas metering station north of unit 1. A review of available current information shows no new pipelines have been installed in the vicinity of Turkey Point, and so the conclusions of the IPEEE remain valid. The potential for a pipeline accident is evaluated as a nearby facility accident (explosion), dispositioned above.	Other than the units 1 and 2 natural gas pipeline evaluated as a nearby facility accident above, there are no other pipelines in sufficient proximity to the plant site to cause a significant hazard. It is therefore concluded that no unique PRA model for pipeline accidents is required in order to assess configuration risk for the RICT Program.

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
Release of Chemicals Stored at the Site	The accidental release of toxic chemicals may affect control room habitability. The initiating event would be a chemical spill or tank rupture caused, for example, by a handling accident, container failure, or some other accident. After the material is released, to contribute significantly to risk, it must be carried by some mechanism to the control room air intake. Evaluations of the very small quantities and locations chlorine, sulfuric acid, and hydrazine (stored on site) demonstrate that concentrations remain acceptably low after a postulated accident. Other chemicals stored on site are screened as not able to cause a challenge to control room habitability.	There are no chemicals on site which can cause a significant challenge to control room habitability, and there is no impact to other plant mitigating equipment. It is therefore concluded that no unique PRA model for chemical releases is required in order to assess configuration risk for the RICT Program.
Seismic Events	Due to the plant being located in a region of very low seismicity, a seismic PRA was not developed for Turkey Point; however, a bounding estimate of the seismic CDF was calculated by integrating the latest seismic hazard curves for Turkey Point with a plant-level fragility curve. The plant-level fragility represents the plant as a whole rather than the component-level fragilities used in a full seismic PRA. The plant-level fragility does not take credit	Seismic risk is not a significant contributor to configuration risk calculations of the RICT Program for Turkey Point. Given that 6.98E-7/year is a bounding estimate, the seismic CDF is low enough that it can be ignored for the 4b application.

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
	for any of the components that are designed to withstand earthquakes whose severity is greater than the HCLPF of 0.15g. Therefore, the seismic CDF calculated in this manner is conservative. The seismic CDF obtained by integrating the Turkey-Point-specific seismic hazard curve from the recent EPRI study (Reference 9) and the plant-level seismic fragility curve from the GI-199 (Reference 10) calculations resulted in a seismic CDF estimate of 6.98E-07 per year.	
Transportation Accidents	The nearest major highways are U. S. 1 and the Florida Turnpike, approximately 8 miles west, which is sufficient to preclude adverse effects on the plant. There are no railroad lines within 5 miles of the plant, which is sufficient to preclude adverse effects on the plant. There are no major transportations lines near the plant. The SRP criteria for transportation accidents are therefore satisfied. Based on a review of satellite images, there are no new transportation routes in the vicinity of Turkey Point, and so the conclusions of the IPEEE remain valid.	Transportation accidents cannot cause damage to the plant, consistent with the SRP criteria. It is therefore concluded that no unique PRA model for transportation accidents is required in order to assess configuration risk for the RICT Program.
Turbine-Generated	From the FSAR Chapter 5 (Reference 8), turbine	The Turkey Point turbine design has an acceptably

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
Missiles	missiles are not considered a potential threat to plant vital systems, structures or components. The originally installed built-up low pressure turbine rotors were replaced with fully integral rotors. This replacement resulted in a reduction of the probability of unacceptable damage due to a turbine missile to within NRC accepted guidelines. Furthermore, the results of an updated probabilistic analysis for turbine valve test frequency performed in 2003 show that probability of turbine missile ejection is within NRC acceptance criteria with turbine stop and control valve testing being performed on a six month frequency. As part of the Unit 3 and Unit 4 extended power uprate, the turbine controls were upgraded to use an electro-hydraulic control system. An updated analysis demonstrated a reduction in the probability of turbine missile generation with unacceptable damage, and so remains within NRC accepted guidelines.	low probability of destructive missile generation. Therefore, no unique PRA model for turbine missiles is required in order to assess configuration risk for the RICT Program.

References

1. ML071200238, *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 (TAC No. MD4995)*, Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.

2. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines*, Revision 0, November 2006.

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. American Society of Mechanical Engineers and American Nuclear Society, *Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME/ANS RA-Sa-2009, New York (NY), February 2009.

6. FP&L Letter L-94-157, *Generic Letter 88-20 Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*, June 24, 1994.

7. NUREG-75/087, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, 1975.

8. Updated Final Safety Analysis Report, Chapter 5, Appendix 5E, *Missile Protection Criteria*, July 14, 2005.

9. Lettis Consultants International (LCI) Project Report Turkey Point Seismic Hazard and Screening Report, .Calculation of Seismic Hazards for CEUS Sites, Project No. 1041, Nicholas Graehl, February 14, 2014.

10. Generic Issue 199 (GI-199) Implications of Updated Probabilistic Seismic Hazard Estimates in Eastern United States on Existing Plants Safety/Risk Assessment, August 2010.

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Enclosure 5

BASELINE CDF AND LERF

Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

Section 4.0, Item 6 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "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.)

This enclosure demonstrates that the total CDF and total LERF are below the guidance of RG 1.174, specifically, 1E-4/year CDF and 1E-5/year LERF, such that the risk metrics of NEI 06-09 may be applied to the Turkey Point Risk-Informed Completion Time (RICT) Program.

Table 5-1 provides the CDF and LERF values that resulted from a quantification of the baseline internal events PRA models (Reference 5), internal flooding analysis (Reference 6), seismic risk estimate (Reference 7), and the internal fire PRA models (Reference 8). Other external hazards are below accepted screening criteria, and therefore do not contribute significantly to the totals.

Table 5-1: Total Baseline Average Annual CDF/ LERF

	Unit 3		Unit 4	
Hazard	**CDF (per rx-yr)**	**LERF (per rx-yr)**	**CDF (per rx-yr)**	**LERF (per rx-yr)**
Internal Events	7.18E-07	1.87E-08	7.13E-07	1.81E-08
Internal Flooding	1.62E-07	8.36E-10	1.13E-07	4.11E-10
Seismic*	6.98E-07	6.98E-08	6.98E-07	6.98E-08
Fire	8.66E-05	5.35E-06	7.69E-05	4.85E-06
Total	8.82E-05	5.37E-06	7.84E-05	4.87E-06

* Seismic LERF estimated at one-tenth seismic CDF

As demonstrated in the table, the total CDF and total LERF are within the guidance of RG 1.174 to permit small changes in risk which may occur during RICT Program implementation of extended Completion Times. Therefore, the Turkey Point RICT Program is consistent with NEI 06-09 guidance.

References

1. ML071200238, 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 (TAC No. MD4995)," Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.

2. NEI 06-09, Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Revision 0, November 2006.

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. PTN-BFJR-00-001, Revision 10, PTN PSA Model Update – EPU, July 16, 2012.

6. PTN-BFJR-11-009, Revision 1, Turkey Point Internal Flooding Analysis, January 29, 2014.

7. PTN-BFJR-14-009, Revision 0, Turkey Point Seismic CDF Estimate, June 2, 2014.

8. Turkey Point Nuclear Plant FPRA Summary Report, NUREG/CR-6850 Task 16, Revision 10, ERIN, July 2014.

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Enclosure 6

JUSTIFICATION OF APPLICATION OF AT-POWER PRA MODELS TO SHUTDOWN MODES

This enclosure is not applicable to the Turkey Point submittal. FPL is proposing to apply the Risk-Informed Completion Time Program only in Modes 1 and 2 and not in the shutdown Modes.

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Enclosure 7

PRA MODEL UPDATE PROCESS

1.0 INTRODUCTION

Section 4.0, Item 8 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "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 EN-AA-105, *Probabilistic Risk Assessment (PRA) Program* (Reference 3), and EN-AA-105-1000, *PRA Configuration Control and Model Maintenance* (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 Configuration Control Program, an interim 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 performed no less frequently than every five years.

2.0 PRA MODEL UPDATE PROCESS

2.1 Internal Event, Internal Flood, Fire, and Seismic Event PRA 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 NextEra/FPL units. The PRA configuration control process delineates the responsibilities and guidelines for updating the full power internal event, internal flood, fire, and seismic PRA models, and includes both periodic and interim 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.

2.2 Review of Plant Changes for Incorporation into the PRA Model

1. Plant changes or discovered conditions, as defined in the PRA Configuration Control Program, are reviewed for potential impact to the PRA models and 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 the PRA configuration control program (including consideration of the cumulative impact of other pending changes) will be immediately 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. Alternative analyses to conservatively bound the expected risk impact of the change will be performed. In such a case, these alternative 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.

3.0 REFERENCES

1. ML071200238, 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 (TAC No. MD4995)," Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.

2. NEI 06-09, Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Revision 0, November 2006.

3. EN-AA-105, *Probabilistic Risk Assessment (PRA) Program.*

4. EN-AA-105-1000, *PRA Configuration Control and Model Maintenance.*

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Enclosure 8

ATTRIBUTES OF THE CRMP MODEL

1.0 Introduction

Section 4.0, Item 9 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "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 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. The scope of structures, systems, and components (SSCs) within the CRMP will be provided. 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. Additional considerations regarding the fire PRA model to address implementation of National Fire Protection Association (NFPA)-805 as the licensing basis for the fire protection program are also discussed at the end of this attachment.

2.0 Translation of Baseline PRA Model for Use in CRMP

The baseline PRA models for internal events, including the internal flood models, are the peer-reviewed models, updated when necessary to incorporate plant changes to reflect the as-built/as-operated plant. The NFPA-805 internal fire models will be used in the RICT Program, and are also updated when necessary to incorporate plant changes to reflect the as-built/as-operated plant. Prior to implementation of the RICT Program, the internal events model will be integrated with the internal flood model and the internal fire model to develop a one-top integrated baseline model. This baseline model will be modified to create the CRMP model to be used for the RICT Program by removing mutually exclusive maintenance events logic excluding configurations prohibited by plant procedures or guidelines, and modifying flag files to allow user-specified train alignments.

The EPRI EOOS software will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. The integrated models may include additional changes that are currently logged in the database for periodic model maintenance and update that are considered pending for the upcoming cycles of model update in accordance with plant procedures.

The current Turkey Point 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.

Development of the integrated models, and the changes that might be required thereof are controlled using plant procedures and calculations, which include all necessary quality controls and reviews.

3.0 Scope of Systems, Structures, and Components within the CRMP

In addition to the SSCs modeled for each TS LCO in the scope of the RICT Program (described in Enclosure 1), the additional SSCs and/or corresponding functions which are in the PRA models but not in plant TS are listed below.

- Instrument Air System and Turbine Cooling Water System
- Main Feedwater and Condensate systems, pumps, and valves.

4.0 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. 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 to verify proper translation of the baseline PRA models and acceptance of all changes made to the baseline PRA models pursuant to translation to the CRMP model. This testing also verifies correct mapping of plant components to the basic events in the CRMP model.

5.0 Training and Qualification

The PRA staff is responsible for development and maintenance of the CRMP model. The PRA staff is trained in accordance with the site's Engineering personnel training program. Operations and Work Control staff will use the CRMP tool under the RICT Program. They are trained in accordance with a program using National Academy for Nuclear Training (ACAD) documents, which is also accredited by INPO.

6.0 Application of the CRMP Tool to the RICT Program Scope

The EPRI EOOS software will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. This program is specifically designed by EPRI to support implementation of RMTS, and is currently used at the site. EOOS will permit the user to evaluate all configurations within the scope of the RICT Program using appropriate mapping of equipment to PRA basic events.

7.0 Additional Considerations for NFPA-805 Modifications

The existing fire PRA model includes credit for committed plant modifications to be implemented as part of the transition of the fire protection licensing basis to NFPA-805 (as described in Reference 5). At the expected time of implementation of the RICT Program, not all of these committed modifications will be implemented. FPL proposes to use the risk insights from the post-transition fire PRA model and commits as part of the RICT Program implementation to maintain compensatory measures in place until the associated plant modifications are implemented, as described in Table S-2 of Attachment S of Reference 5.

8.0 REFERENCES

1. ML071200238, *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 (TAC No. MD4995)*," Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.
2. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4B: Risk-Managed Technical Specifications (RMTS) Guidelines*, Nuclear Energy Institute, Revision 0-A, November 2006.
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. ASME/ANS RA-Sa-2009, *Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008*, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.
5. FPL Letter L-2013-099, "*License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance-Based for Fire Protection for Light Water Reactor Generating Plants (2001) Edition)*," March 22, 2013.

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Enclosure 9

KEY ASSUMPTIONS AND SOURCES OF UNCERTAINTY

Introduction:

Section 4.0, item 10 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "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 how the key assumptions and sources of uncertainty were identified, and how their impact was assessed and dispositioned.

This enclosure provides a discussion of how the key assumptions and sources of uncertainty were identified, and how their impact on the Risk-Informed Completion Time (RICT) Program was assessed and dispositioned.

Process for Identification of Key Assumptions and Sources of Uncertainty:

Sources of model uncertainty and related assumptions, defined consistent with Regulatory Guide (RG) 1.200 Revision 2 (Reference 3) and the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Probabilistic Risk Assessment (PRA) Standard (Reference 4), have been identified for the baseline PRA models using the guidance of NUREG-1855 (Reference 5) and EPRI TR-1016737 *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessment* (Reference 6).

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 and Section 3.1.1 of EPRI TR-1016737. The process in these references was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

Disposition of Key Assumptions and Sources of Uncertainty

The list of assumptions and sources of uncertainty from Reference 7 were reviewed to identify those which would be significant for the evaluation of configuration-specific changes in risk. If the model uses a non-conservative treatment, or methods which are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine the impact on RICT Program calculations. Only those assumptions or sources of uncertainty which could significantly impact the configuration risk calculations were considered key for this application.

The internal events PRA models are used to support the fire and seismic PRA, and so the assumptions and uncertainties evaluated would apply to these PRA models as well.

Key assumptions and sources of uncertainty for the RICT Program application are identified and dispositioned in Table E9-1.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC IE-A-4 Human-induced initiating events. Support system initiating events do not explicitly consider human-induced events.	While not explicitly considered, the generic and plant-specific data used to develop the frequencies of these initiators would include events whose cause was human error.	For those systems in the scope of the RICT Program which are also in the scope of initiating events, risk management actions (RMAs) will include consideration of actions to enhance protection of the remaining available train or buses.
GENERIC IE-A-9 Common cause failures (CCF). CCFs for electrical buses and panels are not included in initiating event models for electrical buses.	CCFs for electrical buses and panels are considered to be very rare events, and PWROG guidelines do not include recommendations for CCF groupings for buses.	For RICT Program delta risk calculations, CCF is not significant since the failure probability is based on the remaining available train or component. NEI 06-09 guidelines require the use of RMAs to address potential CCFs when emergent failures occur. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Loss of 4kv Busses. A reactor trip without a pressure challenge of the pressurizer power-operated relief valves (PORVs) is assumed to occur if bus 3C or 4C is lost; in most cases, a reactor trip would not occur. If a trip did occur at beginning-of-life in the cycle, a PORV challenge might occur, which is not considered.	The overall impact of these assumptions is judged to be slightly conservative, since a reactor trip is being assumed when it most likely would not occur.	The conservative assumptions would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Excessive LOCA mitigation. Core damage is assumed for interfacing systems LOCA (ISLOCA) and reactor vessel rupture initiating events - no mitigation is credited.	The current modeling approach is consistent with current industry practice but introduces a very slight conservatism that should not impact the overall results in a significant manner.	The assumption has a potentially non-conservative impact on the calculated RICT for ECCS systems, since they may actually provide some mitigation capability for these events which would not be reflected in the calculations. Since the assumption is consistent with current industry practice and is judged to be a small impact, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Spurious SI. The Bayesian prior for initiating event frequency is based on relay failure data.	Relay data is considered the best available generic data source for this initiating event. This assumption is judged to have a small impact on overall results.	Since the assumption is judged to be a small impact, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Site Loss-of-Offsite Power. All grid-related and weather-related loss of offsite power events are assumed to result in a dual-unit event.	The assumption is judged to introduce a slight conservatism in results, and would have a negligible impact.	This conservative assumption may cause shorter RICTs for equipment which mitigates loss-of-offsite power events; since the impact is conservative, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
IE-C-2 Applicability of generic data. Low pressure piping failure generic data is used.	The failure of piping given an overpressure event is a key factor in the ISLOCA frequency. The use of generic data is consistent with industry practice.	This assumption does not affect RICT calculations, since only the frequency of ISLOCA initiating events is impacted, and these events are assumed not to be mitigated. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
AS-A-3 Thermal-hydraulic codes. The code MAAP 4 is used and although this is a consensus approach, the NRC has recently posed potential issues regarding the code's appropriate application. (Similar item SC-B-6)	MAAP 4 is a consensus approach for PRA analysis at this time; pending resolution of the NRC issues, this may change and would be re-visited as a PRA update or upgrade as part of the normal PRA model maintenance practices.	As a consensus approach, the use of MAAP 4 is acceptable for PRA. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Small-small LOCA assumptions. A pressurizer PORV sticking open or reactor coolant pump (RCP) seal failure will result in a small-small (S1) LOCA.	This assumption increases the risk contribution of the S1 LOCA and is conservative since S1 LOCAs require secondary heat removal. The impact is judged to be small.	This slight conservatism would have a negligible impact on RICT calculations, and therefore no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Vessel head vent leakage. Leakage from reactor vessel head vent valves is negligible.	This assumption is reasonable given that the valves are small and normally closed such that leakage is very unlikely to occur.	This slight non-conservatism is judged to have a negligible impact on RICT calculations, and therefore no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Credit for secondary steam relief. Failure of secondary steam relief is neglected for secondary heat removal for transients.	Since adequate steam relief is provided by the passive relief valves, the assumption that secondary steam relief always succeeds is judged to only be a small non-conservatism. The modeling is consistent with NUREG-1150 analysis for a similar design plant.	Secondary relief valves are not in the scope of the RICT Program. This slight non-conservatism would have a negligible impact on RICT calculations, and therefore no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC No credit for opposite unit feedwater or use of condensate pumps. Secondary cooling does not credit either the opposite unit pumps, nor the use of low pressure condensate.	This assumption adds a very small conservatism to the results for scenarios in which secondary heat removal fails. This assumption is judged to not impact the results significantly due to the existence of multiple heat removal systems that are modeled.	This slight conservatism in secondary cooling modeling would have a negligible impact on RICT calculations, and therefore no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Pressurizer PORV status. Pressurizer PORV will not open before bleed-and-feed initiation criteria are met.	This assumption is consistent with the expected plant response to a loss of secondary heat removal, and might add a slight conservatism since if the PORV opens, no manual operator action would be required.	This slight conservatism in bleed-and-feed modeling would have a negligible impact on RICT calculations, and therefore no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Bleed-and-feed criteria. Two pressurizer PORVs are required for successful bleed-and-feed cooling.	This assumption is judged to add a small amount of conservatism to the results if further analysis would indicate that only one PORV is needed. This assumption should not significantly affect the base case PRA results.	This slight conservatism in bleed-and-feed modeling would conservatively impact RICT calculations for PORVs, therefore no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Steam generator isolation. For steam generator tube rupture (SGTR) events, failure to isolate any of the following constitutes an isolation failure for the affected SG: main steam isolation valve (MSIV), MSIV bypass valve, SG blowdown lines, and auxiliary feedwater (AFW) steam supply. Main Feedwater isolation and bypass isolation valves close on the safety injection signal, and are not evaluated for failure to remain closed.	These assumptions regarding isolation of a ruptured SG are judged to have a very small impact on the analysis results.	Based on the small impact of these assumptions, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Passive piping failure. For most system models, ruptures and other passive piping failures (other than the tanks and heat exchangers) were not modeled.	This assumption is slightly conservative, but the likelihood of a piping passive failure is judged to be small relative to other active system failures.	Based on the small impact of this assumption, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC SI accumulator mission time. A 24-hour mission time for SI accumulators is assumed.	This assumption is conservative since for large and medium LOCAs, the accumulators function is accomplished in much less time. This is judged to be insignificant to results.	This conservative assumption will result in a slightly conservative calculation of RICTs. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC AFW backflow. Backflow through idle AFW pumps is not modeled.	Two failures (failure of one pump and failure of its check valve to remain closed) must occur for backflow to occur; therefore the probability of this failure path is relatively small. However, it is not known with certainty what the backflow volume would be. This assumption is judged to introduce a small non-conservatism into the PRA results.	Based on the small impact of this assumption, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC AFW success criteria. The AFW success criteria assume loss of main Feedwater.	This assumption introduces a small conservatism into the PRA results, as additional time would be available to initiate AFW flow for events with main Feedwater initially available (due to the greater volume of water available in the SG secondary for some of these events).	This conservative assumption will result in a slightly conservative calculation of RICTs. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Seal injection flow diversion. For RCP seal injection, flow diversion through the *-333 valves was not considered	This assumption is reasonable, given the current plant practice of keeping these valves normally closed.	This reasonable assumption will not impact RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Intake Structure Fouling. Fouling is considered as a potential initiator, but is not considered during the mission time that ICW would be required following an event.	This assumption is judged to introduce a small non-conservatism into the PRA results. The likelihood of a blockage event during post-accident response is relatively small; however, such an event could disable both ICW trains.	Although non-conservative, there is no impact on RICT calculations since fouling is a CCF mode which disables both trains, and is not affected by the service status of the ICW trains. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Relief valve failure probabilities. The failure-to-close likelihood of the pressurizer safety valves and PORVs is unaffected by the initiating event.	The more severe conditions imposed by an ATWS fluid discharge may result in an increased possibility of a stuck-open valve, but there is no data to evaluate this assumption.	This assumption is judged not to impact RICT calculations significantly. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Primary pressure control (PPC) pneumatics. Sufficient backup nitrogen gas is available to allow the PPC to perform its functions in the event of a loss of instrument air.	The assumption may adversely impact the availability of pressurizer PORVs on loss of instrument air.	The nitrogen backup is designed to allow 10 minutes of cumulative operation of the PORVs, sufficient for pressure relief and bleed-and-feed operations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Room cooling. Room cooling is not required for the control building and computer rooms.	This non conservative assumption is reasonable given the likelihood that operators would detect and correct such a failure.	Although non-conservative, requiring room cooling and including appropriate recovery actions for alternate cooling is judged to have a negligible impact on the results. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Room cooling. Only four portable fans are assumed available for backup cooling to the DC Equipment and Inverter Room.	The assumption is reasonable based on the plant design and operating procedures, but is slightly conservative.	This slight conservatism is judged to have a negligible impact on the results. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Staggered testing. Non-staggered testing is assumed when the testing scheme cannot be established.	This assumption is slightly conservative, and the overall impact on the results is judged to be small.	This conservative assumption will result in conservative calculations of RICTs. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC CCF error factors. The error factor calculations assume that the alpha factor uncertainty parameters can be transformed to EFs without the introduction of additional uncertainty.	This assumption impacts only parametric uncertainty calculations. The impact is judged to be small.	Parametric uncertainty calculations are not used in the RICT Program. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC CCF error factors The CCF alpha factors are assumed to have an upper bound error factor of 15.	This assumption impacts only parametric uncertainty calculations. The impact is judged to be small.	Parametric uncertainty calculations are not used in the RICT Program. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
Internal flood floor drain impacts. No credit is taken for drain flow in the internal flooding analysis, only flow to other rooms.	This assumption is conservative, but could introduce uncertainty in flood scenario timing. No detailed evaluation of the potential for such impacts is documented.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
Internal flood release rate. The maximum release rate is that occasioned by the guillotine rupture of a line.	This assumption is expected to conservatively impact internal flooding calculations. However, flood mitigation features may have different relative importance based on the flood size.	It is judged that this conservative assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
Internal flood spray protection. Instruments and electrical panels which are environmentally qualified for exposure to steam are able to withstand the effects of spray or splashing. Equipment further than 30 feet from a spray source would not be damaged. (Similar assumption IF-C-13)	This assumption is based on engineering judgment.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
Internal flood equipment damage. Equipment damaged by flooding will remain unavailable for the duration of the flood.	This assumption is conservative in that repair and recovery of equipment is not credited.	This assumption is consistent with NEI 06-09 guidelines which do not permit crediting repairs to damaged equipment. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
Internal flood reactor trip. The plant is assumed to be tripped if a flood might lead the operators to elect to shut down the plant, or if the flood could require a TS plant shutdown.	This assumption may be slightly conservative since a controlled plant shutdown should be less significant than a reactor trip.	It is judged that this conservative assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
Internal flood mitigation. The operators will not intervene to terminate a flood until 24 hours has elapsed.	This is a conservative assumption.	It is judged that this conservative assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
Internal flood door failure. Doors fail at a flood height of 3-feet.	This assumption could result in changes to flood scenarios and therefore result in conservative or non-conservative changes in results. If the flood involves a large source, then the assumption would not be relevant, since if the door would eventually fail as the flood level continued to increase.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
Internal flood door gaps. A one-quarter inch door gap is assumed unless specific measurements are made.	This assumption is used to estimate flow of water through room doors that are assumed to have a minimum of 1/4" gap unless a specific bigger/smaller gap is measured as indicated.	It is judged that this assumption is reasonable and would not impact RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Grid Stability. Recently the stability of at least some local areas of the electric power grid has been questioned. The potential duration and complexities of recovery from such events may not be reflected in the offsite power recovery analysis.	Loss of offsite power frequency and recovery are based on industry-wide data and plant-specific battery capabilities to permit alignment of breakers for recovery.	There is no statistical basis to assess loss of offsite power events differently at this time. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC Support system initiating events. The use of plant-specific models has led to inconsistencies in the treatment of CCF and equipment recovery.	Recovery of support system initiators is not credited. A mean time to repair of a failed train is assessed to determine the appropriate mission time for the second train, where applicable for a support system initiator. CCF is treated in the mitigating system tree.	Not crediting recovery for support system initiators is conservative for estimating the frequency of these initiating events, which conservatively impacts RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Operation of equipment after battery depletion. Some PRAs have credited manual operation of equipment after battery depletion for station blackout events.	No credit is taken for operation of AC independent systems after battery depletion. It is assumed that operation of systems following battery depletion is not possible.	Not crediting manual operation of equipment is conservative, which conservatively impacts RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Core cooling success following containment failure. Loss of containment heat removal leading to long-term containment over-pressurization and failure can be a significant contributor in some PRAs. Consideration of the containment failure mode might result in additional mechanical failures of credited systems. Containment venting through "soft" ducts or containment failure can result in loss of core cooling due to environmental impacts on equipment in the reactor building, loss of NPSH on ECCS pumps, steam binding of ECCS pumps, or damage to injection piping or valves.	Success of containment heat removal is required to support success of ECCS recirculation.	This source of uncertainty is addressed by requiring containment heat removal to support ECCS recirculation. No additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC Containment sump/strainer performance. All PWRs are improving ECCS sump management practices, including installation of new sump strainers at most plants. There is not a consistent method for the treatment of ECCS sump performance.	The risk of containment sump plugging has been significantly decreased due to a complete sump re-design. Sump clogging is not modeled in the PRA.	This source of uncertainty is slightly non-conservative, but reasonable given the improved design of the containment sump. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Impact of failure of RCS pressure relief. Certain scenarios can lead to RCS/RPV pressure transients requiring pressure relief. Usually, there is sufficient capacity to accommodate the pressure transient. However, in some scenarios, failure of adequate pressure relief can be a consideration. Various assumptions can be taken on the impact of inadequate pressure relief.	Generic success criteria based on CEOG guidance for pressure relief are used. Failure of pressure relief is assumed to proceed to core damage.	Since the failure mode is addressed in the PRA consistent with applicable owner group guidance, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Operability of equipment in beyond design basis environments. Due to the scope of PRAs, scenarios may arise where equipment is exposed to beyond design basis environments (w/o room cooling, w/o component cooling, w/ deadheading, in the presence of an un-isolated LOCA in the area, etc.)	It is assumed that equipment will fail to operate in conditions beyond its environmental qualifications. This assumption may add a small amount of conservatism to the base PRA results, if equipment could continue to operate under these conditions. However, the impact of this assumption on the PRA results is believed to be small.	This assumption conservatively treats the source of uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC Credit for Emergency Response Organization (ERO). Most PRAs do not give much, if any credit, for initiation of the ERO, including actions included in plant-specific severe accident management guidelines (SAMGs) and the new B5b mitigation strategies. The additional resources and capabilities brought to bear via the ERO can be substantial, especially for long-term events.	Credit for ERO is not taken. Credit for some direction from the ERO for longer-term actions would be a realistic assumption, and not crediting these actions is a slight conservative treatment.	This assumption conservatively treats the source of uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Core melt arrest in-vessel. Typically, the treatment of core melt arrest in-vessel has been limited. However, recent NRC work has indicated that there may be more potential than previously credited.	Arresting an in-vessel core melt event is only included for loss of offsite power sequences through recovery of offsite power. All SBO sequences that do not arrest core melt progression are assumed to have no CHR capability.	This assumption conservatively treats the source of uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Ex-vessel cooling of lower head. The lower vessel head of some plants may be submerged in water prior to the relocation of core debris to the lower head. This presents the potential for the core debris to be retained in-vessel by ex-vessel cooling. This is a complex analysis impacted by insulation, vessel design and degree of submergence.	Ex-vessel cooling of the lower head is not considered due to uncertainties in the behavior of the lower head penetrations and the presence of insulation surrounding the lower head. This is considered a realistic treatment.	This assumption realistically treats the source of uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC Core debris contact with containment. In some plants, core debris can come in contact with the containment shell (e.g., some BWR Mark Is, some PWRs including free-standing steel containments). Molten core debris can challenge the integrity of the containment boundary. Some analyses have demonstrated that core debris can be cooled by overlying water pools.	This is not considered as an early failure mechanism because there is no direct path for core debris to contact the containment shell.	This assumption realistically treats the source of uncertainty with regards to early containment failure, and therefore there is no impact on LERF calculations for a RICT. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

References

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2. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document*, Nuclear Energy Institute, Revision 0, November 2006.
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.
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5. NUREG-1855, Volume 1, *Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making.*
6. EPRI TR-1016737, *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments*, December 2008.
7. PTN-PRA-SNBK-UNCERT, *Uncertainty Analysis Notebook Turkey Point Units 3 and 4*, April 2014.
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Enclosure 10

PROGRAM IMPLEMENTATION

Introduction

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 risk management action implementation during extended Completion Times (CT).

RICT Program and Procedures

NextEra 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 Risk Managed Technical Specifications included in NEI 06-09.

The Operations Department (licensed operators) is responsible for compliance with the Technical Specifications (TS) and will be responsible for implementation of RICTs and risk management actions (RMA). 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 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 PRA functionality.
- 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.

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 NextEra personnel:

Site Personnel

- Operations Director
- Operations Personnel (Licensed and Non- Licensed)
- Operations Training
- Outage Manager
- On-line Manager
- Planning and Scheduling Personnel
- Work Week Managers
- Licensing Personnel
- Selected Maintenance Personnel
- Engineering
- Risk Engineering
- Other Selected Management

Corporate Personnel

- Operations Corporate Functional Area Manager
- Fleet Outages Corporate Functional Area Manager
- Licensing Management and Personnel
- Risk Engineering Management and Personnel
- Training Management and Personnel
- Other Selected Management

Training will be carried out in accordance with NextEra 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. NextEra 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

Level 2 Training

This training is applicable to supervisors, managers, and other personnel who need a broad understanding of the RICT Program. It 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 and case studies are not included. The concepts of the RICT Program will be taught, but this group of personnel will not be qualified to perform the tasks for actual implementation of the RICT Program.

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.

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Enclosure 11

MONITORING PROGRAM

Introduction

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.

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. 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, 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), 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

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,* 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.

Turkey Point Nuclear Plant, Units 3 and 4

Enclosure 12

RISK MANAGEMENT ACTION EXAMPLES

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.

Responsibilities

For planned entries into the RICT Program, the department responsible for performing the maintenance or other activity is responsible for developing the RMAs with assistance from Operations and the Probabilistic Risk Assessment Group (PRAG). Operations is responsible for approval and implementation of RMAs. For emergent entry into extended CTs, Operations is also responsible for developing the RMAs.

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. The RMAs will be implemented at the earliest possible time, without waiting for the actual RMAT to 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 re-evaluation 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 are implemented 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. 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 (10 CFR 50.65 (a)(4) Fire) associated with that SSC shall be implemented per the current plant procedure.

It is possible to credit RMAs in RICT calculations; however, such quantification of RMAs is neither required nor expected by NEI 06-09. Nonetheless, if RMAs will be credited to determine realistic 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 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

Examples

Example RMAs that may be considered during a RICT Program entry for a diesel generator (DG) or a battery to reduce the risk impact and ensure adequate defense-in-depth are:

A. Diesel Generator:

1. Evaluate the condition of the offsite power supply, switchyard, and the grid prior to entering a RICT, and implement the RMAs below during times of high grid stress conditions, such as during high demand conditions.
2. Defer switchyard activities, such as of discretionary maintenance on the main, auxiliary, or startup transformers associated with the unit.
3. Defer maintenance that affects the reliability of the trains associated with the operable DGs.
4. Defer planned maintenance activities on station blackout mitigating systems, and treat those systems as protected equipment.
5. Contact the dispatcher on a periodic basis to provide information on DG status and the power needs of the facility, and to obtain grid status.
6. Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the impacted DG.

B. Battery:

1. Limit the immediate discharge of the affected battery, if possible.
2. Recharge the affected battery to float voltage conditions using a spare battery charger, if possible.
3. Evaluate the capacity of the remaining battery and protect its ability to perform its safety function.
4. Periodically verify battery float voltage is equal to or greater than the minimum required float voltage for remaining batteries.