Southern Nuclear

Regulatory Affairs

40 Inverness Center Parkway
Post Office Box 1295
Birmingham, AL 35242

205 992 5000 tel
205 992 7601 fax

July 27, 2018

Docket Nos.: 50-348
50-364

NL-18-0039

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

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines"

Ladies and Gentlemen:

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Southern Nuclear Operating Company (SNC) is submitting a request for an amendment to the Technical Specifications for the Joseph M. Farley Nuclear Plant (FNP).

The proposed amendment would modify TS requirements to permit use of Risk Informed Completion Times in accordance with NEI 06-09, Revision 0-A, *Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines*.

- Attachment 1 provides a description and assessment of the proposed changes, 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 (Provided for Information Only)

SNC submits this change as a site-specific application. Technical Specification Task Force (TSTF)-505 is not utilized for this application, as the most recent version of this traveler is not approved. On March 12, 2018, SNC staff discussed this application with NRC staff. This application is consistent with the guidelines of NEI 06-09, Revision 0-A and the Risk Informed Technical Specifications Program approved by The Nuclear Regulatory Commission (NRC) for SNC's Vogtle Electric Generating Plant (VEGP), Units 1 and 2, on August 8, 2017, CAC NOS. ME9555 and ME 9556.

SNC requests approval of the proposed license amendment by August 1, 2019 with the amendment being implemented within 120 days of issuance..

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

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 Alabama Official.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 27 day of July 2018.

Respectfully submitted,

Cheryl A. Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/PDB/SCM

Attachments:

1. Basis for Proposed Change
2. Marked-Up Technical Specification Changes
3. Clean Typed Technical Specification Changes
4. Marked-Up Technical Specification Bases

Enclosures:

1. List of Revised Required Actions to Corresponding PRA Functions.
2. Information Supporting Consistency with Regulatory Guide 1.200, Revision 2.
3. Information Supporting Justification of Bounding Analysis or Excluding Sources of Risk Not Addressed by the PRA Models
4. Baseline CDF and LERF
5. PRA Model Update Process
6. Attributes of the CRMP Model
7. Key Assumptions and Sources of Uncertainty
8. Program Implementation
9. Risk and Performance Monitoring Program
10. Risk Management Action Examples

cc: Regional Administrator, Region II
NRR Project Manager – Farley
Senior Resident Inspector – Farley
Director, Alabama Office of Radiation Control
RTYPE: CFA04.054

Joseph M. Farley Nuclear Plant- Units 1&2

License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines"

Attachment 1

Basis for Proposed Change

1. Summary Description

The proposed amendment would modify the Farley Nuclear Plant (FNP) Technical Specification (TS) requirements related to completion times (CTs) for required actions (RAs) to provide the option to calculate a longer, risk-informed completion time (RICT). The allowance is described in a new program in Chapter 5, "Administrative Controls," entitled the "Risk Informed Completion Time Program."

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

The proposed amendment is consistent with the methodologies presented in TSTF-505, Revision 1, *Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b*. Although the proposed amendment is consistent with TSTF-505, SNC is not proposing adoption of TSTF-505 with this License Amendment Request (LAR). This LAR is a site-specific application. Only those required actions described in this attachment and Enclosure 1 are proposed to be changed. This is consistent with the methodology described in NEI 06-09, Revision 0-A.

2. Detailed Description

The proposed amendment would modify the FNP TSs in the following manner to incorporate the RICT Program.

Use and Application Example 1.3-8, which demonstrates the format and use of the RICT Program within a limiting condition of operation (LCO), is added to the TS and reads as follows:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. ---- NOTES ---- 1. Not applicable when second subsystem intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------ Two subsystems inoperable.	B.1 Restore subsystems to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

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

If a second subsystem is declared inoperable, Condition B may also be entered. The Condition is modified by two Notes. The first note states it is not applicable if the second subsystem is intentionally made inoperable. The second note provides restrictions applicable to these “loss of function” Conditions. The Required Actions of Condition B are not intended for voluntary removal of redundant subsystems from service. The Required Action is only applicable if one subsystem is inoperable for any reason and the second subsystem is found to be inoperable, or if both subsystems are found to be inoperable at the same time. If Condition B is applicable, at least one subsystem must be restored to OPERABLE status within 1 hour or Condition C must also be entered. The licensee may be able to apply a RICT or to extend the Completion Time beyond 1 hour, but not longer than 24 hours, if the requirements of the Risk Informed Completion Time Program are met. If two subsystems are inoperable and Condition B is not applicable (i.e., the second subsystem was intentionally made inoperable), LCO 3.0.3 is entered as there is no applicable Condition.

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

If the 7 day Completion Time clock of Condition A or the 1 hour Completion Time clock of Condition B have expired and subsequent changes in plant conditions result in exiting the applicability of the Risk Informed Completion Time Program without restoring the inoperable subsystem to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start.

If the RICT expires or is recalculated to be less than the elapsed time since the Condition was entered and the inoperable subsystem has not been restored to OPERABLE status, Condition C is also entered and the Completion Time clocks for

Required Actions C.1 and C.2 start. If the inoperable subsystems are restored to OPERABLE status after Condition C is entered, Conditions A, B, and C are exited, and therefore, the Required Actions of Condition C may be terminated.

Administrative Controls Section 5.5.20, which describes the RICT Program, is added to TSs and reads as follows. This is consistent with TSTF-505 and NEI 06-09, Revision 0-A and amended for the adjustments made to the Vogtle Electric Generating Plant (VEGP) Risk Informed TS Program during NRC review:

Risk Informed Completion Time Program

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

a. The RICT may not exceed 30 days.

b. A RICT may only be used in MODE 1 and 2.

c. When a RICT is being used, any plant change within the scope of the Configuration Risk Management 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 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 of more of the trains are considered "PRA Functional" as defined in Section 2.3.1 of NEI 06-09, Revision 0-A. The RICT for these loss of function conditions may not exceed 24 hours

f. 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, Revision 0-A. However, the following additional constraints shall be applied to the criteria for “PRA Functional”:

1. Any structures, systems and components (SSCs) credited on the PRA Functionality determination shall be the same SSC relied upon to perform the specified safety Technical Specifications safety function.
2. Design basis success criteria parameters shall be met for all design basis accident scenarios for establishing PRA Functionality during a Technical Specifications loss of function (LOF) condition where a RICT is applied.

g. Use of a RICT for LOF conditions may not exceed 24 hours. Upon entering a RICT, the potential for common cause failure (CCF) must be addressed. This can be accomplished in one of two ways:

1. Adjusting the common cause factors in the configuration risk management tool,

   OR

2. Implementing risk management actions (RMA) which specifically address the potential for the CCF. If RMAs are chosen as the method for addressing the potential for the CCF, those RMAs must be in effect prior to reaching the front stop.

   If it is determined that a CCF is not likely, the RMAs or common cause adjustment factors may be discontinued.

h. A RICT entry is not permitted, or a RICT entry made shall be exited, for any condition involving a TS loss of function if a PRA Functionality determination that reflects the plant configuration concludes that the LCO cannot be restored without placing the TS inoperable trains in an alignment which results in a loss of functional level PRA success.

Individual LCO Required Actions (RA) modified by the proposed amendment to be included in the RICT program are identified below. Notes regarding of TSTF-505 refer to TSTF-505-A, Rev. 1. In many cases, new Conditions were added. In the descriptions below, the letter of the Condition refers to the new designation, not the previous letter designation. In some cases, TSTF-505 may include additional Actions for which FNP is not requesting approval. Only the Actions proposed to be modified are discussed.

3.4.10 Pressurizer Safety Valves

Required Action A.1 – Restore valve to OPERABLE status

- Condition A: One pressurizer safety valve inoperable).
- FNP is proposing an option to calculate a RICT and LOF designation for LCO 3.4.10, Action A.1 which is consistent with the VEGP safety evaluation.

- This deviates from the TSTF-505 LCO Condition in that a LOF Condition is assigned to Condition A of the FNP LCO. This is because the FNP safety analysis assumes operation of all three pressurizer safety valves to limit increases in RCS pressure.

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

Required Action B.3 – Restore PORV to OPERABLE status

- Condition B: One PORV inoperable and not capable of being manually cycled)
- FNP is proposing an option to calculate a RICT for LCO 3.4.11 Action B.3 which is consistent with the VEGP SE and TSTF-505.

Required Action C.2 – Restore block valve to OPERABLE status

- Condition C: One block valve inoperable)
- FNP is proposing an option to calculate a RICT for LCO 3.4.11 Action C.2 which is consistent with the VEGP SE and TSTF-505.

Required Action F.2 – Restore one block valve to OPERABLE status

- Condition F: Two block valves inoperable; this is a LOF Condition.
- FNP is proposing this option to calculate a RICT for LCO 3.4.11 Action F.2 which is consistent with the VEGP SE (Action F.1, in VEGP TSs). VEGP was approved as a LOF Condition. This LCO deviates from TSTF-505 in the following manner: Condition F is a LOF Condition in the FNP TS; it is not in the corresponding TSTF-505 Condition. Condition F in the FNP TS differs from the corresponding Condition in the NUREG in that there are two Required Actions in the FNP Condition as opposed to one in the TSTF; consequently, a RICT is assigned to Required Action F.2.

3.5.1 Accumulators

Required Action C.1 – Restore one accumulator to OPERABLE status.

- Condition C: Two or more accumulators inoperable for reasons other than boron concentration not within limits.)
- FNP is proposing an option to calculate a RICT for LCO 3.5.1 Action C.1 which is consistent with the VEGP SE and TSTF-505. TSTF-505 identifies this Condition as a LOF. VEGP was also approved as a LOF Condition. This

Condition was renumbered from Condition D to Condition C and an hour was given for the completion time where previously 3.0.3 entry was immediately required.

3.5.2 ECCS – Operating

Required Action A.1 – Restore train(s) to OPERABLE status

- Condition A: One or more trains inoperable AND at least 100% of the ECCS flow equivalent to a single Operable ECCS train available
- FNP is proposing an option to calculate a RICT for LCO 3.5.2 Action A.1 which is consistent with the VEGP SE and TSTF-505.

3.5.4 Refueling Water Storage Tank (RWST)

Out of date Notes are being deleted for this LCO. These Notes specifically indicated they cannot be used after Spring of 2015 (U1) and 2016 (U2).

Required Action B.1 – Restore RWST to OPERABLE status

- Condition B: RWST inoperable for reasons other than Condition A, this is a LOF condition.
- FNP is proposing an option to calculate a RICT for LCO 3.5.4 Action B.1 which is consistent with the VEGP SE approval (Action E.1, in VEGP TSs). TSTF-505 does not identify this Condition as a LOF. VEGP was approved as a LOF Condition because with the RWST inoperable, neither the ECCS nor the Containment Spray system can perform its design function.

3.6.2 Containment Air Locks

Required Action C.3 – Restore air lock to OPERABLE status

- Condition C: One or more containment air locks inoperable for reasons other than Condition A or B
- FNP is proposing an option to calculate a RICT for LCO 3.6.2 Action C.3 which is consistent with the VEGP SE and TSTF-505.

3.6.3 Containment Isolation Valves

Required Action A.1 – Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

- Condition A: One or more penetration flow paths with one containment isolation valve inoperable except for purge valve penetration leakage not within limit. Note: Only applicable to penetration flow paths with two containment isolation valves.
- FNP is proposing an option to calculate a RICT for LCO 3.6.3 Action A.1 which is consistent with the VEGP SE and TSTF-505.

Required Action B.1 – Isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange.

- Condition B: One or more penetration flow paths with two containment isolation valves inoperable except for purge valve penetration leakage not within limit. Note: Only applicable to penetration flow paths with two containment isolation valves. This is a LOF Condition.
- FNP is proposing an option to calculate a RICT for LCO 3.6.3 Action B.1 which is consistent with the VEGP SE approval. TSTF-505 does not identify this Condition as a LOF.

Required Action C.1 – Isolate the affected flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.

- Condition C – One or more penetration flow paths with one containment isolation valve inoperable. Note: Only applicable to penetration flow paths with only one containment isolation valve and a closed system. This is a LOF Condition.
- FNP is proposing an option to calculate a RICT for LCO 3.6.3 Action C.1 which is consistent with the VEGP SE approval. TSTF-505 does not identify this Condition as a LOF.

3.6.6 Containment Spray and Cooling Systems

Required Action A.1 – Restore containment spray train to OPERABLE status.

- Condition A: One containment spray train inoperable
- FNP is proposing an option to calculate a RICT for LCO 3.6.6 Action A.1 which is consistent with the VEGP SE and TSTF-505.

Required Action B.1 – Restore one containment spray train to OPERABLE status.

- Condition B: Two containment spray trains inoperable. This is a TS LOF Condition.
- This change was not approved in the VEGP SE; however, this Condition was documented in TSTF-505 as a LOF Condition (Condition E.1 in TSTF-505).

Required Action D.1 Restore containment cooling train to OPERABLE status.

- Condition D: One containment cooling train inoperable).
- FNP is proposing an option to calculate a RICT for LCO 3.6.6 Action D.1 which is consistent with the VEGP SE (Action B.1 in VEGP TSs) and TSTF-505 (Action C.1 in TSTF-505).

Required Action E.1 Restore one containment cooling train to OPERABLE status.

- Condition E: Two containment cooling trains inoperable
- This change was not approved in the VEGP SE; however, this Condition was documented in TSTF-505. (Condition D.1 in TSTF-505).

Required Action G.1 – Restore one containment spray or cooling train to OPERABLE status.

- Condition G: Any combination of three or more trains inoperable. This is a LOF Condition.
- This change was not approved in the VEGP SE; however, this Condition was documented in TSTF-505 as a LOF Condition (Condition E.1 in TSTF-505).
- FNP LCO 3.6.6 deviates in format from the TSTF-505 mark-up but the effect is the same. FNP has added Condition B, “Two containment spray trains inoperable”, and Condition G is “Any combination of three or more trains inoperable”. Both are LOF Conditions. SNC FNP proposes separating the *two containment spray trains inoperable* Condition from the *any combination of three or more trains inoperable* Condition because it is clearer and more concise than the TSTF-505 Condition G where both situations are considered in one TS Condition.

3.7.2 Main Steam Isolation Valves (MSIVs)

Required Action A.1 – Restore MSIV to OPERABLE status.

- Condition A: One or more steam lines with one MSIV inoperable in MODE 1.

- FNP is proposing an option to calculate a RICT for LCO 3.7.2 Action B.1 which is consistent with the VEGP SE and TSTF-505.

Required Action B.1- Restore one MSIV to OPERABLE status in affected steam line.

- Condition B: One or more steam lines with two MSIVs inoperable in MODE 1. This is a TS LOF Condition.
- FNP is proposing an option to calculate a RICT for LCO 3.7.2 Action B.1 which is consistent with the VEGP SE and TSTF-505.

3.7.4 Atmospheric Relief Valves (ARVs)

Required Action A.1 – Restore required ARV line to OPERABLE status.

- Condition A: One required ARV line inoperable.
- FNP is proposing an option to calculate a RICT for LCO 3.7.4 Action A.1, which is consistent with TSTF-505.
- Vogtle did not request approval for this action because the existing Completion time is already 30 days.

Conditions B and C are deviations from the TSTF and from VEGP. For the purposes of the Risk Informed Completion Time Program, the Condition of “Two or more required ARV lines inoperable” is being split into two Conditions. Condition B will be “Two required ARV lines inoperable” and Condition C will be “Three required ARV lines inoperable”. Condition C is a LOF Condition, but Condition B is not. These conditions were separated because otherwise Condition B would have been a LOF with only 2 ARVs inoperable.

Required Action B.1 – Restore one ARV line to OPERABLE status.

- Condition B: Two required ARV lines inoperable).
- FNP is proposing an option to calculate a RICT for LCO 3.7.4 Action B.1 which is similar to TSTF-505 (Condition B.1 is “Two or more”).
- For VEGP, this new Condition was not proposed.

Required Action C.1 – Restore one ARV line to OPERABLE status.

- Condition C: Three required ARV lines inoperable. This is a LOF Condition.
- FNP is proposing an option to calculate a RICT for LCO 3.7.4 Action C.1 which is similar to the VEGP SE approval. VEGP Condition B.1 is *Two or more required ARV lines inoperable*. VEGP Action B.1 was approved as a LOF

Condition. TSTF-505 does not identify this Condition as a LOF.

3.7.5 Auxiliary Feedwater System (AFW)

Required Action A.1 – Restore affected equipment to OPERABLE status.

- Condition A: One steam supply to turbine driven AFW pump inoperable.
- FNP is proposing an option to calculate a RICT for LCO 3.7.5 Action A.1 which is consistent with the VEGP SE and TSTF-505.

Required Action B.1 – Restore AFW train to OPERABLE status.

- Condition B: One AFW train inoperable for reasons other than Condition A.
- FNP is proposing an option to calculate a RICT for LCO 3.7.5 Action B.1 which is consistent with the VEGP SE and TSTF-505.

3.7.6 Condensate Storage Tank (CST)

Required Action A.2 – Restore CST to OPERABLE status.

- Condition A: CST inoperable. This is a LOF condition.
- Although requested consistent with TSTF-505 guidance, an option to calculate a RICT for LCO 3.7.5 was not approved in the VEGP SE. This is a deviation from TSTF-505 in that Condition A, “CST Inoperable”, is identified as a LOF Condition in the FNP proposed TS. This is because the CST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis. If this source of water is unavailable, it may not be possible to mitigate these events.

3.7.7 Component Cooling Water (CCW) System

Required Action A.1 – Restore CCW train to OPERABLE status

- Condition A – One CCW train inoperable
- FNP is proposing an option to calculate a RICT for LCO 3.7.7 Action A.1 which is consistent with the VEGP SE and TSTF-505.

Required Action B.1 – Restore one CCW train to OPERABLE status.

- Condition B: Two CCW trains inoperable. Condition B is a new Condition, and has been added as a LOF.

- FNP is proposing an option to calculate a RICT for LCO 3.7.7 Action B.1 as a LOF, which is consistent with the VEGP SE and TSTF-505.

3.7.8 Service Water System (SWS)

Required Action A.1 – Restore SWS train to OPERABLE status.

- Condition A: One SWS train inoperable
- FNP is proposing an option to calculate a RICT for LCO 3.7.8 Action A.1 which is consistent with the VEGP SE and TSTF-505.

Required Action B.1 – Restore one SWS train to OPERABLE status.

- Condition B: Two SWS trains inoperable. Condition B is a new Condition, added as a TS LOF.
- FNP is proposing an option to calculate a RICT for LCO 3.7.8 new Action B.1 as a LOF, which is consistent with the VEGP SE and TSTF-505.

3.7.11 Control Room Air Conditioning System (CRACS)

Required Action E.1 – Restore one CRACS train to OPERABLE status.

- Condition E: Two CRACS trains inoperable in MODE1, 2, 3, or 4. This is a LOF Condition.
- FNP is proposing an option to calculate a RICT for LCO 3.7.11 Action E.1 which is consistent with the TSTF-505.
- This LCO was not considered for the VEGP RMTS program.

3.7.19 Engineered Safety Feature (ESF) Room Coolers

Required Action A.1: - Restore the affected ESF Room Cooler subsystem Train to OPERABLE status.

- Condition A: One required ESF Room Cooler subsystem Train inoperable.

Required Action B.1: - Restore one of the same ESF Room Cooler subsystems to OPERABLE status.

- Condition B: Two trains of the same ESF Room Cooler subsystem inoperable.

This LCO does not exist in NUREG-1431. Consequently, this represents a deviation from TSTF-505. VEGP received approval for LCO 3.7.14 (Actions A.1 & B.1 in VEGP SE), which are very similar to LCO 3.7.19 for FNP. Consistent with VEGP, Condition B is considered a LOF Condition for FNP.

3.8.1 AC Sources – Operating

Required Action A.3 – Restore required offsite circuit to OPERABLE status.

- Condition A: One required offsite circuit inoperable)
- FNP is proposing an option to calculate a RICT for LCO 3.8.1 Action A.3 which is consistent with the VEGP SE and TSTF-505.

Required Action B.4 – Restore DG set to OPERABLE status.

- Condition B: One DG Set inoperable)
- FNP is proposing an option to calculate a RICT for LCO 3.8.1 Action B.4 which is consistent with the VEGP SE and TSTF-505.

Required Action C.2 – Restore one required offsite circuit to OPERABLE status.

- Condition C: Two required offsite circuits inoperable
- FNP is proposing an option to calculate a RICT for LCO 3.8.1 Action C.2 which is consistent with the VEGP SE and TSTF-505.

Required Action D.1 – Restore required offsite circuit to OPERABLE status.
Required Action D.2 – Restore DG set to OPERABLE status.

- Condition D: One required offsite circuit inoperable AND One DG set inoperable).
- FNP is proposing an option to calculate a RICT for LCO 3.8.1 Actions D.1 and D.2 which is consistent with the VEGP SE and TSTF-505.

Required Action E.1 – Restore one DG set to OPERABLE status.

- Condition E: Two DG Sets inoperable)
- FNP is proposing an option to calculate a RICT for LCO 3.8.1 Actions E.1 which is consistent with the VEGP SE and TSTF-505.
- Condition E includes a deviation from TSTF-505. This is due to the structure of the LCO Condition differing between the FNP TS and the Standard TS, marked up for TSTF-505. The FNP Condition is two DG sets inoperable. There are three Completion Times in the current FNP Required Action but only one in the TSTF and VEGP SE. The FNP

CTs are dependent on which combination of individual DGs is affected. The CT increases depending on the severity in the combinations of DGs that are inoperable. The first combination listed in the current FNP CT for Condition E is 2 hours for all three DGs inoperable. The next two are 8 hours and 24 hours for different inoperable combinations of 2 DGs. A RICT is being assigned to the 8 hour and 24 hour CT. The first CT is being eliminated because it will be covered in proposed Condition H.

Required Action G.1 – Restore automatic load sequencer to OPERABLE status.

- Condition G: One automatic load sequencer inoperable)
- FNP is proposing an option to calculate a RICT for LCO 3.8.1 Action G.1 which is consistent with the VEGP SE and TSTF-505.

Required Action H.1 – Restore one required AC source to OPERABLE status.

- Condition H: Three or more required AC sources inoperable).
- FNP is proposing an option to calculate a RICT for LCO 3.8.1 Actions H.1 which is consistent with the VEGP SE (Action is G.1 in VEGP) and TSTF-505 (Action is G.1 in TSTF). This is considered a LOF Condition. This was previously Condition I. The previous Required Action was immediate 3.0.3 entry. In this proposal, an hour is given to perform a RICT calculation.

3.8.4 DC Sources – Operating

Neither VEGP nor FNP have adopted TSTF-500; therefore, all Conditions of this LCO deviate from TSTF-505 in that NUREG-1431, which was used as the generic mark-up for the Risk Informed Tech Specs, incorporates TSTF-500, “DC Electrical Re-write”.

Required Action A.1 – Restore the Auxiliary Building DC electrical power subsystem to OPERABLE status.

- Condition A: One auxiliary building DC electrical power subsystem inoperable).

- FNP is proposing an option to calculate a RICT for LCO 3.8.4 Action A.1 which does not entirely align with the VEGP SE and TSTF-505.

- Condition A is slightly different for VEGP, *one DC electrical power source inoperable due inoperable battery A or B*.

- An outdated option for Action A.1 is being deleted. The option was only applicable for Cycle 19.

The remaining Actions do not perfectly align with VEGP due to plant design differences, but the intent is similar.

Required B.1 – Restore the battery connection resistance to within limit.

- Condition B: One auxiliary building DC electrical power subsystem with battery connection resistance not within limit.

Required Action D.1- Restore the battery connection resistance to within the limit.

- Condition D: One required SWIS DC electrical power subsystem battery connection resistance not within limit.

Required Action F.1 – Restore at least one DC electrical power subsystem to Operable status.

- Condition F: Two or more DC electrical power subsystems inoperable that result in a LOF. This is a new Condition, added as a LOF

FNP is proposing options to calculate a RICT for LCO 3.8.4 Actions B.1, D.1, and F.1 as LOF Conditions. As indicated above, these do not align with TSTF-505 and existing Condition descriptions are different from Vogtle Conditions due to naming conventions and design differences, but the intent is similar to the VEGP SE.

3.8.7 Inverters – Operating

Required Action A.1 – Restore inverter to OPERABLE status.

- Condition A: One required inverter inoperable).
- FNP is proposing an option to calculate a RICT for LCO 3.8.7 Action A.1 which is consistent with the VEGP SE and TSTF-505.

Required Action B.1 – Restore one required inverter to OPERABLE status.

- Condition B: Two or more required inverters inoperable. This is a new Condition and is added as a LOF).

- FNP is proposing an option to calculate a RICT for LCO 3.8.7 Action B.1. This Condition is considered a LOF which is consistent with the VEGP SE and TSTF-505.

3.8.9 Distribution Systems – Operating

Required Action D.1 – Restore AC electrical power distribution subsystem(s) to OPERABLE status.

- Condition D: One or more AC electrical power distribution subsystems inoperable.
- FNP is proposing an option to calculate a RICT for LCO 3.8.9 Action D.1. This is consistent with the VEGP SE (Condition A.1 in VEGP) and TSTF-505.

Required Action E.1 – Restore AC Vital bus subsystem(s) to OPERABLE status.

- Condition E: One or more AC Vital buses inoperable).
- FNP is proposing an option to calculate a RICT for LCO 3.8.9 Action E.1. This is consistent with the VEGP SE (Condition B.1 in VEGP) and TSTF-505.

Required Action F.1 – Restore auxiliary building DC electrical power distribution subsystem to OPERABLE status.

- Condition F: One auxiliary building DC electrical power distribution subsystem inoperable).
- FNP is proposing an option to calculate a RICT for LCO 3.8.9 Action F.1. This is similar to VEGP SE (Condition C.1 in VEGP) and TSTF-505.

Required Action G.1 – Restore one train to OPERABLE status.

- Condition G: Two trains with inoperable distribution subsystems that result in a loss of safety function. This is a LOF Condition.

- FNP is proposing an option to calculate a RICT for LCO 3.8.9 Action G.1. This is similar to VEGP SE (Condition D.1 in VEGP) and TSTF-505.

3. Technical Evaluation

The proposed modification to FNP Units 1 and 2 TS would add Section 5.5.20, Risk Informed Completion Time (RICT) Program to Chapter 5, Administrative Controls, add Example 1.3-8 to Chapter 1, Use of Application, and modify selected Required Action (RA) Completion Times (CT), provided risk is assessed and managed as described in NEI 06-09, Revision 0-A. In accordance with NEI 06-09, Revision 0-A, PRA methods are used to justify each extension to a RA CT based on the specific plant configuration, which exists at the time of the applicability of the RA, and are updated when plant configurations change. This application includes documentation regarding the technical adequacy of the PRA models used in the Configuration Risk Management Program (CRMP), consistent with the requirements of RG 1.200 (Enclosure 2).

Most TS LCOs identify one or more Conditions for which the LCO may not be met, to permit a licensee to perform required testing, maintenance, or repair activities. Each Condition has associated RAs for restoration of the LCO or for other actions, each with some fixed time interval, referred to as the Completion Time, which identifies the time interval permitted to complete the Required Action. Upon expiration of the CT, the licensee is required to shut down the reactor or follow the remedial action(s) stated in the TS. The RICT program provides the necessary administrative controls to permit extension of CTs and thereby delay reactor shutdown or remedial actions, if risk is assessed and managed within specified limits and programmatic requirements. The specified safety function of performance levels of TS required SSCs are unchanged, and the remedial actions, including the requirement to shut down the reactor, are also unchanged; only the CTs for the RAs are extended by the RICT program.

NEI 06-09, Revision 0-A allows the application of a RICT to emergent conditions which represent 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” as defined in Section 2.3.1 of NEI 06-09, Revision 0-A. In order to avoid intentional entry into these “loss of function” conditions, they are modified by a Note similar to: “Not applicable when the second system [train] [division] is intentionally made inoperable”. A second Note, added to these loss of function (LOF) conditions, lists the restrictions on these conditions, as given in Section 5.5.20. Furthermore, any SSCs credited in the PRA Functional determination shall be the same SSCs relied upon to perform the specified Technical Specifications safety function and design basis parameters will be met.

The Bases for each specific LOF Condition are expanded to discuss the Note, similar to:

“The Condition is modified by two Notes. The first Note stating it is not applicable when the second system [train] [division] is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one system [train] [division] is inoperable for any reason and a second system [train] [division] is found to be inoperable, or if two systems [trains] [divisions] are found to be inoperable at the same time. The second Note lists the restrictions, per TS Section 5.5.20, that are applicable to these LOF conditions”.

In Section 4, “Limitations and Conditions”, of the Safety Evaluation for NEI 06-09, Revision 0-A, there are thirteen aspects listed that describe required, plant-specific information to support a license amendment request to adopt NEI 06-09, Revision 0-A. They are as follows:

(1) The LAR will include proposed changes to the administrative controls of TS to add a Configuration Risk Management Program (CRMP) in accordance with NEI 06-09-A, Revision 0-A.

This information can be found in Attachment 1.

(2) The LAR will provide identification of the TS LCOs and Action requirements to which the Risk Managed Technical Specifications (RMTS) will apply, with a comparison of the TS functions to the PRA modeled functions of the 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, flowrate, etc., are consistent licensing basis assumptions (i.e., 50.46 ECCS flowrates) for each of the TS requirements, or an appropriate disposition or programmatic restriction will be provided.

This information can be found in Enclosure 1.

(3) The LAR will provide a discussion of the results of peer reviews and self-assessments conducted for the plant-specific PRA models which support the RMTS, including the resolution or disposition of any identified deficiencies (i.e., findings and observations from peer reviews). This will include a comparison of the requirements of RG 1.200 using the elements of ASME RA-Sb-2005 for capability Category II for internal events PRA models, and for other models for which RG 1.200 endorsed standards exist. If additional standards have been endorsed by revision to RG 1.200, the LAR will provide similar information for those PRA models used to support the RMTS program.

This information can be found in Enclosure 2.

(4) The LAR will provide a description, in terms of scope, level of detail, technical adequacy, and methods applied, for all PRA models used in calculations of risk used to support the RMTS for risk sources for which NRC endorsed standards are not available.

This item is not applicable to this license amendment request.

(5) The LAR will provide a justification for excluding any risk sources determined to be insignificant to the calculation of configuration –specific risk, and will provide a discussion of any conservative or bounding analysis to be applied to the calculation of RICTs for sources of risk not addressed by the PRA models.

This information can be found in Enclosure 3.

(6) The LAR will provide the plant-specific total CDF and total LERF to confirm that these are less than 10^{-4}/year and 10^{-5}/year, respectively.

This information can be found in Enclosure 4.

(7) This assures that the potential risk increases allowed under the RMTS program are consistent with RG 1.174, Revision 3.

The information can be found in Enclosure 4.

(8) The LAR will provide appropriate plant-specific justification for using at-power PRA models in shutdown modes to which the RMTS applies.

This item is not applicable to this license amendment request.

(9) The LAR will 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 information can be found in Enclosure 5.

(10) The LAR will provide a description of the PRA models and tools used to support the RMTS, including identification of how the baseline 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 SSCs within the CRMP will be provided. This item should also confirm that the CRMP tools can be readily applied for each TS LCO within the scope of the plant-specific RMTS submittal

This information can be found in Enclosure 6.

(11) The LAR will provide a discussion of how the key assumptions and sources of uncertainty were identified, and how their impact on the RMTS was assessed and dispositioned.

This information can be found in Enclosure 7

(12) The LAR will provide a description of the implementing programs and procedures regarding the plant staff responsibilities for the RMTS implementation, and specifically discuss the decision process for RMA implementation during a RICT.

This information can be found in Enclosure 8

(13) The LAR will include a description of the implementation and monitoring program as described in RG 1.174, Revision 3, Section 2.3, Element 3, and TR NEI 06-09, Revision 0-A, Section 2.3.2, Step 7.

This information can be found in Enclosure 9.

(14) The LAR will describe the process to identify and provide compensatory measures and RMAs during expected CTs. Provide examples of compensatory measures/RMAs for planned activities which exceed risk levels identified in NUMARC 93-01 (RMA threshold) that involve an extended CT.

This information can be found in Enclosure 10.

4. Summary of Vogtle Electric Generating Plant (VEGP) Responses to Requests for Additional Information

This section provides a summary of selected responses to NRC requests for additional information received by SNC during the VEGP Risk Informed Completion Time (RICT) program review process. These summaries are with respect to how they pertain to the FNP RICT Program. Those responses in which commitments were made by SNC are included in this section. Reviews and confirmations which were made, for FNP, as a result of the VEGP RAI responses are also included. Those VEGP responses which were only applicable to VEGP are not included. Also, not included are those responses which only provided clarification on existing SNC practices, procedures, and processes.

In each case, only the relevant portions of the NRC question are provided. However, the SNC RAI response letter number and the date of the letter are included in each case should reviewers want to see the entire VEGP RAI response for the particular question.

In general, any response to a VEGP RAI which discusses SNC fleet procedures, processes, and guidelines pertaining to the Risk Informed Completion Time Program and makes clarifications regarding those processes, were not included in this section. It is understood that the clarifications made in the VEGP submittal regarding these general items will apply to the FNP Risk Informed Completion Time Program as well. These fleet procedures will be made applicable to both sites when FNP receives approval; therefore, those procedure clarifications will also be applied to FNP.

The following SNC responses are provided as they pertain to FNP.

1) <u>NRC Question #4, from SNC letter NL-13-1540, August 2, 2013</u>

"… Please address how the VEGP updated final safety analysis report will be revised to reflect the new conditions and required actions."

<u>SNC Response for FNP:</u>

SNC will include a summary of the Risk Informed Completion Time Program in the FNP FSAR. This will include a section on PRA Functionality which will list those conditions which must be satisfied before declaring a component as "PRA Functional" per the NEI 06-09, Revision 0-A guidelines. The section will explicitly state that for a TS component to be considered PRA Functional, its PRA success criteria, among other things, must be satisfied. Additionally, for loss of LOF, the SSCs' design basis criteria for parameters must also be satisfied.

The FNP FSAR discussion will also include a section on PRA adequacy. It will state that the on-record PRA model that forms the basis for the VEGP Configuration Risk Management (CRM) tool has been developed to the requirements of Reg Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", and is subjected to peer reviews per the requirements of NRC endorsed PRA standards and SNC procedures. Those peer reviews are formally documented along with the findings and observations of the review and their corresponding resolutions.

2) NRC Question #5, from SNC letter NL-14-1016, dated July 17, 2014

An oversight occurred during the NRC review of TSTF-505, Revision 1, and a specific scenario was not satisfactorily addressed. SNC is requested to address the following scenario.

For this scenario, the TS system is comprised of train A and train B and performs two associated Probabilistic Risk Assessment (PRA) success criteria, called PRA function 1 and PRA function 2.

In an emergent condition, with both TS system train A and train B TS inoperable and the associated PRA success criteria considered PRA functional with train A able to perform PRA function 1 and train B being able to perform function 2 (i.e., neither train by itself can perform PRA functions 1 and 2 but both trains together maintain PRA functionality). The NEI 06-09, Revision 0-A guidelines will allow a risk informed completion time to be entered in this scenario, however, there is no way to repair either train A or train B without losing PRA functionality.

…Please provide changes to the proposed “Risk Informed Completion Time Program,” in VEGP TS 5.5.22, which prevents entry into a risk informed completion time for this specific scenario.

SNC Response for FNP:

The following statement will be placed in new Section 5.5.20 of the FNP TS:

“A RICT entry is not permitted, or a RICT entry made shall be exited, for any condition involving a TSLOF if a PRA Functionality determination that reflects the plant configuration concludes that the LCO cannot be restored without placing the TS inoperable trains in an alignment which results in a loss of functional level PRA success criteria.”

3) NRC Question #13, from SNC letter NL-14-1016, dated July 17, 2014.

“… In a number of instances, the disposition in Table E1.1 justifies such differences as PRA success criteria representing “more realistic success criteria.” Since the PRA success criteria differ in some instances from design basis criteria, please confirm that the PRA success criteria is up-to- date, clearly and fully documented for the “4b” application to the level of detail necessary for the RICT program, and appropriate review processes are being implemented for the supporting calculations.

SNC Response for FNP:

Success criteria are documented as part of the PRA documentation and included in the scope of the peer review. PRA success criteria for each system included in the scope of the RICT program are further documented in the “CRM System Guidelines: including flow rates, where applicable, for ease of use during PRA Functionality evaluations when a RICT is entered.

The PRA success criteria are documented in a SNC calculation, which is governed by SNC procedures. The success criteria calculations are living documents and are maintained to reflect the as-built, as-operated plant conditions. SNC calculations are performed by qualified individuals and include a preparer, a reviewer, and an approver.

Table E1.1 of this letter documents the TS LCO Conditions included in the scope of the FNP RICT Program for a comparison between the design basis and PRA success criteria. It also documents, in the “Disposition Column” of Table E1.1, a satisfactory disposition where a difference was identified. Since all differences, as documented in Table E1.1 were satisfactorily resolved, no programmatic restrictions were necessary.

4) <u>NRC RAI #1 (Alternative SSCs), from SNC letter NL-16-0067, dated February 17, 2016</u>

If a PRA Functional determination for a loss of specified safety function or inoperability of all required trains or divisions of a system credits SSCs other than the SSCs covered by TSs (e.g., crediting the Fire Protection system as an alternative water source), please summarize each such TS and justify how appropriate redundancy and diversity is maintained if alternative SSCs are credited.

<u>SNC Response for FNP:</u>

SSCs credited in the PRA Functionality determination are the same SSCs relied on to perform the specified safety function when a RICT for a TS LOF Condition is calculated.

If SNC desires to credit specific alternative SSCs in the future, i.e., SSCs other than those covered by the TS, a separate license amendment request will be required.

5) <u>NRC RAI #1 (Human Actions) from SNC letter NL-16-0067, dated February 16, 2016.</u>

Please confirm that all human action required to achieve PRA functional upon loss of specified safety function are modeled in the PRA (i.e., are proceduralized and trained on or are simple enough so as to be skill of the craft). If any action were evaluated not modeled, please summarize the actions and the evaluation.

<u>SNC Response for FNP:</u>

Human actions required to achieve PRA Functionality during a TS LOF Condition are modeled in the PRA and are proceduralized and trained on unless they are simple enough to be skill of the craft.

6) <u>NRC RAI #1 (Intent of Design Basis) from SNC letter NL-16-0067, dated February 16, 2016</u>

Please confirm that PRA Functionality does not include any scenarios that allow any design basis accident to proceed directly to core damage or containment failure.

SNC Response for FNP:

When in a TS LOF RICT, PRA Functionality determination will include a review of dominant internal events CDF and LERF cutsets to provide high confidence that none of

the design basis accidents, as modeled in the internal events PRA, proceed directly to core damage or containment failure.

7) NRC RAI #2 (SSCs not supporting CDF/LERF) from SNC letter NL-16-0067, dated February 16, 2016

Please confirm that the acceptable PRA Functional modelled in the PRA is also available and sufficient for the remaining design basis accident scenarios that are not modelled in the PRA because other design basis accident scenario does not affect CDF or LERF.

SNC Response for FNP:

For design basis initiators that are not modeled in the PRA because they do not affect CDF or LERF, the PRA Functionality evaluation performed following a TS LOF Condition will ensure SSCs not supporting CDF/LERF will remain available and will sufficiently perform their safety function with respect to the credited design basis scenario.

8) NRC Question #2 (Design Basis Success Criteria) from SNC letter NL-16-0067, dated February 16, 2016.

In Table E1.1 of its application dated September 13, 2012, the licensee noted differences between the design basis success criteria and the PRA success criteria for certain specified safety functions.

…please elaborate on how adequate safety margins are maintained and provide some clarifying examples of adequate safety margins for where the PRA success criteria (e.g., flow rates, temperature limits) differ from the design criteria.

SNC Response for FNP:

For design basis initiators modelled in the internal events PRA, PRA Functionality determination performed subsequent to a TS LOF Condition entry will ensure design basis success criteria for parameters (e.g., flow rates, temperature limits) are met.

9) NRC Question (VEGP LCO 3.8.1, “AC Sources – Operating”) from SNC letter NL-16-0307, dated April 18, 2016.

The LAR proposes to add the option of either applying the existing front stop Completion Time or applying a Risk Informed Completion Time for Required Action C.1. The proposed change to the Completion Time for Required Action C.1 could permit operation for an extended period of time with one DG inoperable without verifying the availability of the SAT or of the CTG. Please provide technical justification, including a discussion of defense-in-depth and safety margin considerations, for the addition of a risk informed completion time for the Required Actions associated with LCO 3.8.1 Condition C, or propose a modification to the license amendment request that retains the existing CTs for verifying availability of SAT and functionality of a DG.

SNC Response for FNP:

The VEGP TS, prior to the approval and implementation of the Risk Informed Completion Time Program into the current TS, contained a risk-informed LCO 3.8.1 which allowed a

14-day Completion Time (CT) for one inoperable diesel generator provided the availability of Start-Up Transformer (SUT) and a Combustion Turbine Generator (CTG) could be confirmed. Ultimately, the LCO 3.8.1 section of the VEGP TS for the Risk Informed Completion Time program was revised to reflect LCO 3.8.1 of the NUREG-1431 standard and TSTF-505. Accordingly, the front stop CT for the DG was changed from 14 days to 72 hours, per the standard.

FNP LCO 3.8.1 currently has a 10 day CT. This is not a risk-informed completion time, in other words, the original justification for the 10 day CT was not risk-informed. Therefore, SNC proposes that the front stop remain at 10 days, with the option of calculating a RICT.

10) NRC Question PRA RAI S-1 (A) from SNC letter NL-16-1008, dated July 13, 2016.

The NRC staff requests SNC to discuss the completion times backstop associated with TS-LOF and its basis. In particular the NRC requests SNC to clarify whether it intends to adopt a 24-hour backstop (and if so, how it intends to do so, in addition to providing marked up TS pages). And whether SNC intends to revise TS 5.5.22 to incorporate the following constraints delineated SNC's previous response (And if so, how it intends to do so, in addition to providing marked-up TS pages):

i) Alternative SSCs cannot replace the SSCs covered by the TSs as described in the response to RAI 1.a.
ii) Design basis success criteria parameters shall be met for design basis accident scenarios that are not modeled in the internal events PRA as described in the response to 2.a.
iii) Design basis success criteria parameters shall be met for design basis accident scenarios modelled in the internal events PRA as described in the response to 2.c.

SNC Response for FNP:

SNC intends to adopt a 24-hour backstop for LOF conditions in the FNP Risk Informed Completion Time Program.

The three additional constraints listed above will also be adopted by FNP and placed in FNP's proposed corresponding description of the Risk Informed Completion Time Program, Section 5.5.20.

11) NRC Question PRA RAI S-2 from SNC letter NL-16-1008, dated July 13, 2016.

C. The NRC staff requests SNC to identify any proposed changes to the TSs that conflict with the constraints or controls identified in PRA RAI S-1 and to provide a disposition of any conflict.

SNC Response for FNP:

The FNP LAR and the FNP proposed TS changes were prepared with the constraints and controls of PRA RAI S-1 in mind. The FNP LCO Conditions which are proposed to include a risk informed completion time do not conflict with the restrictions of question PRA-RAI S-1 from the NRC review of the VEGP risk informed TS.

12) NRC Question DORL-RAI-1 from SNC letter NL-16-1008, dated July 13, 2016.

… NEI 06-09, Revision 0-A incorporates changes based on the NRC staff's safety evaluation dated May 7, 2007, of NEI 06-09, Revision 0-A in the TS. NRC asks VEGP, if needed, to submit marked-up TS pages that reference Revision 0-A of NEI 06-09, Revision 0-A

SNC Response for FNP:

Although the FNP submittal is a site-specific TS change request, SNC will nonetheless use NEI 06-09, Revision 0-A as the implementation guideline and reference it in proposed Section 5.5.20, "Risk Informed Completion Time Program".

13) NRC Question #2 from SNC letter NL-17-0232, dated March 13, 2017.

… SNC provided a list of systems with descriptions of the TS LOF Conditions. The proposed TS 5.5.22 in the same RAI response contains several constraints (e.g., 24 hour backstop and remaining mitigating capabilities) on developing a RICT that can be used for these conditions. However, the proposed TS changes do not identify the Conditions to which these constraints apply. Please propose a modification to the affected TS that stipulates that Conditions will be subject to the 24 hour backstop and associated mitigating capabilities.

SNC Response for FNP:

Section 5.5.20, "Risk Informed Completion Time Program" will contain general rules for the program. Including those that apply specifically to LOF conditions. Additionally, each individual LOF Condition will reference, in a Note, to those specific parts of 5.5.20 applicable to LOF Conditions.

14) NRC Question #3 from SNC letter NL-17-0232, dated March 13, 2017.

The staff reviewed the proposed TS 5.5.22, Risk Informed Completion Time Program, as provided in Enclosure 3 in the letter dated July 13, 2016, and identified the need for some additional clarification.

(1) Enclosure 3, part c, currently states:

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

The proposed wording appears to be circular. The parallel limitation from the NRC SE on NEI 06-09, Revision 0-A is:

> c. When a RICT is being used, any plant configuration change within the scope of the Configuration Risk Management Program (CRMP) must be considered for the effect on the RICT.

Please clarify the logic of the proposed limitation or revise TS 5.5.22 accordingly.

(2) Enclosure 3, part e.2 and 3.3 currently state:

> e.2 For design basis accident scenarios that are not modelled in the PRA because they do not affect the CDF or LERF, the PRA Functionality evaluation performed following a TS LOF Condition entry will ensure SSCs not supporting CDF/LERF will remain available and sufficient.
>
> e.3 For design basis initiators modeled in the internal events PRA, the PRA Functionality determination performed subsequent to a TS LOF Condition entry will ensure design basis success criteria for parameters (e.g., flow rate, temperature limits) are met.

(NRC further indicated in this question that SNC's proposed words, as presented above, did not match NRC's suggested wording, and that it (SNC's words) "substantively changed the scope of the response". NRC went on to suggest additional alternate wording).

<u>SNC Response to part (1) for FNP:</u>

The applicable portion of FNP Section 5.5.20 will use the same words and phrasing as that from the NRC SE on NEI 06-09, Revision 0-A transcribed above.

<u>SNC Response to part (2) for FNP:</u>

SNC will use the same wording for FNP as for VEGP:

> Design basis success criteria parameters shall be met for all design basis accident scenarios for establishing PRA Functionality where a RICT is applied.

15) <u>NRC Question #7 from SNC letter NL-17-0232, dated March 13, 2017.</u>

LCO 3.5.1.A, "One accumulator inoperable due to boron concentration not within limits", is proposed in the scope of the RICT program. In response to RAI #12 provided in letter dated July 17, 2014, the licensee stated that this condition will be modeled in the PRA by assuming loss of accumulator as a surrogate. The RAI response further states that "loss of accumulator is the worst case surrogate for this degraded condition."

" … a) explain how modeling the accumulator as unavailable (i.e., no injection) in the PRA represents the worst case impact of the accumulator boron concentration not being within limits or remove Condition 3.5.1.A from the RICT program.

b) …

SNC Response for FNP

As was done for the VEGP Program, this LCO Condition will not be included in the FNP Risk Informed Completion Time Program.

NRC Question #11 from SNC letter NL-17-0232, dated March 13, 2017.

Please provided a license condition limiting the scope of the PRA and non-PRA methods to what is approved by the NRC staff for use in the plant specific RMTS program. An example is provided below:

The risk assessment approach and methods shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods currently approved by the NRC for generic use. If a licensee wishes to change its method and the change is outside the bounds of this license condition, the licensee will need prior NRC approval, via license amendment.

SNC Response for FNP

FNP will adopt a similar license condition. Attachment 5 contains the marked-up and clean pages for the operating license with this particular condition included.

16) NRC Question 10.3 from SNC letter NL-17-0447, dated April 14, 2017

The proposed changes to the TS include Condition 3.4.11.F , Two [Pressurizer Power Operated Relief Valve – PORV] Block Valves inoperable. The current TS require restoring one block valve to Operable status within 2 hours. The proposed change is to permit the option of calculating a RICT for this Required Action. Per the proposed RICT program, the RICT could be calculated to be any length of time between 2 hours and 30 days. The TS bases state that an Operable block valve may be either open and energized, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be Operable when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). A TS LOF is considered to exist when two redundant SSCs are simultaneously inoperable. Voluntary entry into a condition representing a TS LOF is prohibited throughout the proposed TSs by a Note which modifies the Condition. If emergent conditions create a TS LOF condition, the RICT is limited to maximum of 24 hours and constraints on PRA Functionality are applied. The required position of the PORV block valves could be either open or closed, dependent on the condition of its associated PORV. If the block valves are not repositionable, then inoperability of the block valves could result in a loss of safety function.

SNC Response for FNP

Similar to the VEGP response, this will be made a LOF condition in the FNP RICT Program.

17) NRC Question #10.4 from SNC letter NL-17-0447, dated April 14, 2017

The proposed changes to the TS include Condition 3.5.1.B, One Accumulator Inoperable (for reasons other than Boron Concentration).

The current TS require restoring the accumulator to Operable status within 24 hours. The proposed change is to permit the option of calculating a RICT for this Required Action. Per the proposed RICT program, the RICT could be calculated to be any length of time between 24 hours and 30 days.

Section 6.3.2 of the Vogtle FSAR states that ECCS components are designed such that a minimum of three accumulators, one residual heat removal pump, one residual heat removal (RHR) heat exchanger, together with their associated valves and piping will ensure adequate core cooling in the event of a design basis accident.

The Vogtle TS Bases states that the need to ensure that three accumulators are adequate for this function is consistent with the loss-of-coolant-accident (LOCA) assumption that the entire contents of one accumulator will be lost via the reactor coolant system (RCS) pipe break during the blowdown phase of the LOCA.

It is not clear to the staff how the assumptions in the accident analysis would be satisfied for a LOCA in which the contents of one accumulator is lost through the break, and a second accumulator is inoperable at the time of the event.

Please provide an explanation of how the PRA functionality would be applied in this condition, why this condition would not be considered a TS LOF, and how it would be assured that design basis success criteria would be satisfied.

SNC Response for FNP

Like the VEGP TS, the LCO Condition for FNP was also revised from a one hour CT to a 24 hour CT. The arguments in support of the amendment were risk informed. Therefore, this LCO Condition (3.5.1.B) will be excluded from the FNP RICT program.

18) NRC Question #10.5 from SNC letter NL-17-0447, dated April 14, 2017

The proposed change to the TS include Condition 3.6.3.B, Containment Penetrations with more than one inoperable containment isolation valve, and Condition 3.6.3.C, Containment Penetrations with Purge Valves Leakage outside limits.

The Required Action for Condition B is to isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. The current Completion Time to isolate the penetration flow path is one hour, which is consistent with the time specified to restore containment leakage to within its limits in TS LCO 3.6.1. Additionally, there is a requirement to verify the affected penetration flow path is isolated for at least 31 days for devices outside containment.

Condition C applies when one or more penetration flow paths have one or more containment purge valves not within purge valve leakage limits. The required action is to isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.

The proposed change is to permit the option of calculating a RICT for these Required Actions. Per the proposed RICT program, the RICT could be calculated to be any length of time between 1 hour, for Condition B, and 24 hours for Condition C and 30 days. During this period, no actions would be required to isolate the affected penetration pathway(s); and automatic actions to isolate the pathway may not be assured.

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. The containment penetrations covered under conditions 3.6.3.B and C include those penetrations that are connected directly to the RCS or to the containment atmosphere, and are typically isolated using two isolation devices in series. If both of the isolation devices are open in the isolated position, the safety function of minimizing the loss of reactor coolant inventory and maintaining the containment pressure boundary would not be assured.

Please provide justification to support extension of the Completion Time up to a maximum of 30 days or remove those conditions from the scope of the RICT program. Please include an explanation of how PRA functionality would be applied in this Condition, why this condition would not be considered a LOF, and how it would be assured that design basis success criteria would be satisfied.

<u>SNC Response for FNP</u>

Conditions B and C, ”One or more penetration flow paths with two containment isolation valves inoperable except for purge valve penetration leakage not within limit”, and “One or more penetration flow paths with one containment isolation valve inoperable”, respectively, will be added to the program as LOF conditions. Condition B is applicable to penetrations with two containment isolation valves and Condition C is applicable to penetrations with one containment isolation valve.

19) <u>NRC Verbal Question #1 from SNC letter NL-17-0447, dated April 14, 2017</u>

In Condition 3.4.11.E. “Two PORVs inoperable and incapable of being manually cycled” requires closing and de-energizing the block valves. The current REQUIRED ACTION (RA) statement for this LCO Condition requires closing the associated block valves and removing their power (RAs E.1 and E.2). FSAR Section 15.5.5.1.2.1 describes the *inadvertent operation of emergency core cooling systems during power operation* (IOECCS) event.

For this event, a manual operator action is assumed to open one PORV for water relief. The safety analysis assumes that the PORV is opened in approximately 10 minutes.

However, if a block valve is closed and de-energized, the time to 1) recover power to the block valve, 2) open the block valve, then 3) open the PORV, may go beyond 10 minutes.

In their verbal request, NRC asked SNC to reconcile the situation.

<u>SNC Response for FNP</u>

As was done for VEGP, this LCO Condition, 3.4.11.E, will be removed from the FNP RICT Program.

20) <u>Common Cause Failure Probabilities</u>

In their requests for additional information letters to SNC of February 3, and March 7, 2017 and subsequent telephone conferences, NRC requested information on the VEGP's proposed handling of potential common cause failures during RICT entries. NRC's questions were answered via SNC letters NL-17-0447 dated April 14, 2017 and NL-17-0783, dated May 4, 2017.

Ultimately, NRC and SNC agreed that common cause failures during RICT entry could be handled either by calculational means or by the implementation of Risk Management Actions specifically intended to mitigate the effects of a common cause failure. Consequently, Paragraph g. was added to Section 5.5.22 of the VEGP TS to describe the means that would be used to mitigate the effects of a common cause failure during RICT entry. The same paragraph will be added to FNP TS Section 5.5.20, as follows:

Upon entering a RICT for an emergent condition, the potential for a common cause (CC) failure must be addressed.
If there is a high degree of confidence, based on the evidence collected, that there is no CC failure mechanism that could affect the redundant components, the RICT calculation may use nominal CC factor probability.
If a high degree of confidence cannot be established that there is no CC failure that could affect redundant components, the RICT shall account for the increased possibility of CC failure. Accounting for the increased possibility of CC failure shall be accomplished by one of two methods. If one of the two methods listed below is not used, the Technical Specifications Front Stop will not be exceeded.

1. *The RICT calculation shall be adjusted to numerically account for the increased possibility of CC failure, in accordance with RG 1.177, as specified in Section A-1.3.2.1 of Appendix A of the RG. Specifically, when a component fails, the CC probability for the remaining redundant components shall be increased to represent the conditional failure probability due to CC failure of these components, in order to account for the possibility, the first failure was caused by a CC mechanism.*

   OR

2. *Prior to exceeding the front stop, RMAs not already credited in the RICT calculation shall be implemented. These RMAs shall target the success of the redundant and/or diverse structures, systems, or components (SSC) of the failed SSC and, if possible, reduce the*

frequency of initiating events which call upon the function(s) performed by the failed SSC. Documentation of RMAs shall be available for NRC review.

SNC Response for FNP

As was done for VEGP, administrative controls discussed above have been incorporated FNP RICT Program, as shown in the TS markups, specifically Section 5.5.20.

5. Regulatory Analysis

5.1 Significant Hazards Evaluation

SNC requests adoption of a change to the Farley Nuclear Plant (FNP) plant-specific technical specifications (TS), to modify the TS requirements related to completion times for required actions to provide the option to calculate a longer, risk-informed completion time. The allowance is described in a new program in Chapter 5, “Administrative Controls”, entitled the “Risk Informed Completion Time Program”.

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

1. 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 risk is assessed and managed within the Risk Informed Completion Time Program. The proposed change does not involve a significant increase in the probability of an accident previously evaluated because the changes involve no change to the plant or its mode 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.

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

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 TS revision does not change the design, configuration, or method of plant operation. The proposed change does not involve a physical alteration of the plant in that 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 within the Risk Informed Completion Time Program. The proposed change implements a risk-informed configuration management program to assure that adequate safety margins 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, SNC 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.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, “Technical Specifications” – 10 CFR 50.36(c)(2) states, “When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the Technical Specifications until the condition can be met.”
The proposed change continues to meet the requirements of this regulation.

10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”, requires monitoring the performance of condition of SSCs against licensee-established goals, in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. Additionally, 10 CFR 50.65(a)(4) requires that assessment and management of the increase in risk that may result from a proposed maintenance activity. The proposed change continues to meet the requirements of this regulation.

This license amendment request is consistent with the guidance set forth in NEI 06-09, Revision 0-A, which was found to be consistent with the guidance set forth in Revision 1 of Chapter 19.0, “Use of Probabilistic Risk Assessment in Plant –Specific, Risk-Informed Decision making: Technical Specifications,” of the Standard Review Plan, NUREG-0800, as well as the guidance of Regulatory Guide (RG) 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis”, and RG 1.177, Revision 0, “An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications”.

RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, establishes requirements for PRA technical adequacy. The PRA supporting the proposed change has been assessed using this regulatory guidance.

5.3 Conclusions

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

6. Environmental Considerations

The proposed TS revision 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 effluent 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.

7. References

1) TSTF-505-A, Rev. 1, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b”, Accession No. ML111650552.
2) Topical Report NEI 06-09, Revision 0-A, “Risk-Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines” Accession No ML12286A322
3) NUREG-0800, Standard Review Plan 19.1, “Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Assessment Results for Risk-Informed Activities”, Revision 3, September 2012, Accession No. ML12193A107.
4) NUREG-0800, Standard Review Plan 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance”, Revision 0, June 2007, Accession No. ML071700658.
5) NUREG-0800, Standard Review Plan 16.1, “Risk-Informed Decision making Technical Specifications”, Revision 1, March 20, Accession No.ML070380228.
6) Regulatory Guide 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis”, January 2018, Accession No. ML17317A256.
7) Regulatory Guide 1.177, Revision 1, “An Approach for Plant-Specific Risk-Informed Decision making: Technical Specifications”, May, 2011, Accession No. ML100910008.
8) Regulatory Guide 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, March, 2009, Accession No. ML090410014.

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines"

Attachment 2

Marked-Up Technical Specifications Pages

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

Insert 1

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

IMMEDIATE COMPLETION TIME

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > the Low Temperature Overpressure Protection (LTOP) System applicability temperature specified in the PTLR.

---NOTE---
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

Insert 2

Insert 3

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 AND B.2	Be in MODE 3. Be in MODE 4 with any RCS cold leg temperatures ≤ the LTOP System applicability temperature specified in the PTLR.	6 hours 12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each PORV and each block valve.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1	Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable and not capable of being manually cycled.	B.1	Close associated block valve.	1 hour
	AND		
	B.2	Remove power from associated block valve.	1 hour
	AND		
	B.3	Restore PORV to OPERABLE status.	72 hours

Insert 3

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One block valve inoperable.	C.1	Place associated PORV in manual control.	1 hour
	<u>AND</u>		
	C.2	Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 4.	12 hours
E. Two PORVs inoperable and not capable of being manually cycled.	E.1	Close associated block valves.	1 hour
	<u>AND</u>		
	E.2	Remove power from associated block valves.	1 hour
	<u>AND</u>		
	E.3	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.4	Be in MODE 4.	12 hours

Insert 3

Insert 4

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Two block valves inoperable.	F.1	Place associated PORVs in manual control.	1 hour
	AND		
	F.2	Restore one block valve to OPERABLE status.	2 hours Insert 3
G. Required Action and associated Completion Time of Condition F not met.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	------------------------------NOTES------------------------------ 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. --	
	Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Three ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

-----NOTE-----
In MODE 3, with RCS pressure > 1000 psig, the accumulators may be inoperable for up to 12 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1	Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1	Restore accumulator to OPERABLE status.	24 hours
Insert 5			
~~C.~~ D ~~Required Action and~~ associated Completion Time of Condition A ~~or B~~ B, or C not met.	~~C~~D.1	Be in MODE 3.	6 hours
	AND		
	~~C~~D.2	Reduce RCS pressure to ≤ 1000 psig.	12 hours
~~D. Two or more accumulators inoperable.~~	~~D.1~~	~~Enter LCO 3.0.3.~~	~~Immediately~~

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS — Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

--NOTES--

1. In MODE 3, the Residual Heat Removal or the Centrifugal Charging Pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

2. Upon entry into MODE 3 from MODE 4, the breaker or disconnect device to the valve operators for MOVs 8706A and 8706B may be locked open for up to 4 hours to allow for repositioning from MODE 4 requirements.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours Insert 3
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	C.1 Enter LCO 3.0.3.	Immediately

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

~~NOTES~~

~~1. RWST piping may be unisolated from non-safety related piping for ≤ 4 hours under administrative controls to perform SR 3.5.4.3.*~~

~~2. RWST piping may be unisolated from non-safety related piping for ≤ 30 days per fuel cycle under administrative controls for filtration or silica removal.*~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
Insert 6		
B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour Insert 3

~~*These Notes can only be applied during the next two fuel Cycles for each Unit. These Notes cannot be used after Refueling Outages 1R26 (Spring 2015) and 2R24 (Spring 2016).~~

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

Insert 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------ Only applicable to penetration flow paths with two containment isolation valves. ------------------------------- One or more penetration flow paths with one containment isolation valve inoperable except for purge valve penetration leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours Insert 3
	<u>AND</u>	
	A.2 ------------NOTE------------ Isolation devices in high radiation areas may be verified by use of administrative means. ------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment

Insert 7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ~~NOTE~~ ~~Only applicable to penetration flow paths with two containment isolation valves.~~ One or more penetration flow paths with two containment isolation valves inoperable except for purge valve penetration leakage not within limit.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ~~NOTE~~ ~~Only applicable to penetration flow paths with only one containment isolation valve and a closed system.~~ One or more penetration flow paths with one containment isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND C.2 ----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. ------------------------------ Verify the affected penetration flow path is isolated.	72 hours Once per 31 days

Insert 3

Insert 8

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable. Insert 9	A.1 Restore containment spray train to OPERABLE status.	72 hours
~~B~~. Required Action and associated Completion Time of Condition A or B not met. C	~~B~~.1 Be in MODE 3. AND ~~B~~.2 ---NOTE--- LCO 3.0.4.a is not applicable when entering MODE 4. --- Be in MODE 4.	6 hours Insert 3 54 hours
~~C~~. One containment cooling train inoperable. D	~~C~~.1 Restore containment cooling train to OPERABLE status.	7 days

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D~~ E D. Two containment cooling trains inoperable.	~~D~~ E D.1 Restore one containment cooling train to OPERABLE status.	72 hours Insert 3
~~E~~ F E. Required Action and associated Completion Time of Condition ~~C~~ D or ~~D~~ E not met. Insert 10	~~E~~ F E.1 Be in MODE 3.	6 hours
	AND	
	~~E~~ F E.2 ---------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 4.	
	Be in MODE 4.	~~12 hours~~ 1 hour
~~F.~~ G ~~Two containment spray trains inoperable.~~	~~F.1~~ G.1 ~~Enter LCO 3.0.3.~~ Restore required trains to OPERABLE status.	~~Immediately~~ Insert 3
~~OR~~		
Any combination of three or more trains inoperable.		

Insert 11

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	---------NOTE--------- Not required to be met for system vent flow paths opened under administrative control.	
	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs per steam line shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when one MSIV in each steam line is closed.

ACTIONS

NOTE
Separate Condition entry is allowed for each steam line.

Insert 3

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam lines with one MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	72 hours
B. One or more steam lines with two MSIVs inoperable in MODE 1.	B.1 Restore one MSIV to OPERABLE status in affected steam line.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours
D. One or more steam lines with one MSIV inoperable in MODE 2 or 3.	D.1 Verify one MSIV closed in affected steam line.	7 days <u>AND</u> Once per 7 days thereafter

Insert 12

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Three ARV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

Insert 13

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ARV line inoperable.	A.1	Restore required ARV line to OPERABLE status.	7 days Insert 3
~~B. Two or more required ARV lines inoperable.~~	~~B.1~~	~~Restore all but one ARV line to OPERABLE status.~~	~~24 hours~~
~~C~~. D Required Action and associated Completion Time of Condition A, B, or C not met.	~~C~~.1 D	Be in MODE 3.	6 hours
	AND		
	~~C~~.2 D	Be in MODE 4.	18 hours

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Turbine driven AFW train inoperable due to one inoperable steam supply. OR ------------NOTE----------- Only applicable if MODE 2 has not been entered following refueling. ------------------------------- One turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days
B. One AFW train inoperable for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours

Insert 3

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

Insert 15

ACTIONS

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

Insert 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST level is ≥ ~~150~~164,000 gal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by CCW. ---------------------------------- Restore CCW train to OPERABLE status.	72 hours Insert 3
Insert 16			
~~B~~ C. Required Action and associated Completion Time of Condition A or B not met.	~~B~~ C.1	Be in MODE 3.	6 hours
	AND		
	~~B~~ C.2	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------- Be in MODE 4.	12 hours

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 **Two SWS trains shall be OPERABLE.**

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SWS train inoperable.	A.1	---NOTES--- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources — Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by SWS. --- Restore SWS train to OPERABLE status.	72 hours~~*~~ Insert 3
Insert 17 ~~B~~. One SWS automatic turbine building isolation valve inoperable in each SWS train.	~~B~~.1 C	Restore both inoperable turbine building isolation valves to OPERABLE status.	72 hours

~~* For the FNP Unit 2 October 06, 2009 entry into Technical Specification 3.7.8, the Service Water Train A may be inoperable for a period not to exceed 7 days provided that during the extended completion time for Train A, two Train A pumps are available (OPERABLE except during a seismic event).~~

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
~~C.~~ D Required Action and associated Completion Time of Condition ~~A or B~~ A, B, or C not met.	~~C~~D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	~~C~~D.2	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------- Be in MODE 4.	 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	-----------------------------NOTE------------------------------ Isolation of SWS flow to individual components does not render the SWS inoperable. --- Verify each accessible SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each SWS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.4	Verify the integrity of the SWS buried piping by visual inspection of the ground area.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two CRACS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	D.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	D.2	Suspend movement of irradiated fuel assemblies.	Immediately
Insert 18			
E. Two CRACS trains inoperable in MODE 1, 2, 3, or 4.	E.1	~~Enter LCO 3.0.3.~~ Insert 19	~~Immediately~~

Insert 20

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CRACS train has the capability to remove the assumed heat load.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.19 Engineered Safety Feature (ESF) Room Coolers

LCO 3.7.19 ESF Room Coolers shall be OPERABLE.

APPLICABILITY: When associated ESF equipment is required to be OPERABLE.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each ESF Room Cooler subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ESF Room Cooler subsystem Train inoperable.	A.1 Restore the affected ESF Room Cooler subsystem Train to OPERABLE status.	72 hours
Insert 21		Insert 3
~~B.~~ C Required Action and associated Completion Time of Condition A or B not met. ~~OR~~ ~~Two trains of the same ESF Room Cooler subsystem inoperable.~~	~~B~~ C.1 Be in MODE 3. AND ~~B~~ C.2 Be in MODE 5.	6 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore required offsite circuit to OPERABLE status.	72 hours Insert 3
B. One DG set inoperable.	---NOTE--- LCO 3.0.4c is applicable when only one of the three DGs is inoperable. ---		
	B.1	Perform SR 3.8.1.1 for the required offsite circuit(s).	2 hours AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable DG set inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE DG set is not inoperable due to common cause failure.	24 hours
	OR		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.2	Perform SR 3.8.1.6 for OPERABLE DG set.	24 hours
	AND		
	B.4	Restore DG set to OPERABLE status.	10 days
C. Two required offsite circuits inoperable. Insert 3	C.1	Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required features
	AND		
	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable. AND One DG set inoperable.	---NOTE--- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems—Operating," when Condition D is entered with no AC power source to any train. ---	
	D.1 Restore required offsite circuit to OPERABLE status.	24 hours
	OR	
	D.2 Restore DG set to OPERABLE status.	24 hours
E. Two DG sets inoperable. Insert 3	E.1 Restore one DG set to OPERABLE status.	~~2 hours if all three DGs are inoperable~~ ~~OR~~ 8 hours if DG 1-2A and DG 1(2)B are inoperable OR 24 hours if DG 1C and DG 1(2)B are inoperable
F. Required Action and associated Completion Time of Condition C or E not met.	F.1 Be in MODE 3.	6 hours

, or H

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. One automatic load sequencer inoperable.	G.1	Restore automatic load sequencer to OPERABLE status.	12 hours
Insert 22			Insert 3
~~H~~ I. Required Action and associated Completion Time of Condition A, B, D, ~~or~~ G not met.	~~H~~ I.1	Be in MODE 3.	6 hours
	AND		
	~~H~~ I.2	---NOTE--- LCO 3.0.4.a is not applicable when entering MODE 4. --- Be in MODE 4.	12 hours
~~I. Three or more required AC sources inoperable.~~	~~I.1~~	~~Enter LCO 3.0.3.~~	~~Immediately~~

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—Operating

LCO 3.8.4 The Train A and Train B Auxiliary Building and Service Water Intake Structure (SWIS) DC electrical power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One Auxiliary Building DC electrical power subsystem inoperable.	A.1	Restore the Auxiliary Building DC electrical power subsystem to OPERABLE status.	2 hours ~~12 hours for 1B Auxiliary Building DC electrical power subsystem inoperable due to inoperable battery for cycle 19 only~~
B. One Auxiliary Building DC electrical power subsystem with battery connection resistance not within limit.	B.1	Restore the battery connection resistance to within limit.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 AND C.2	Be in MODE 3. -----NOTE----- LCO 3.0.4.a is not applicable when entering MODE 4. --------------- Be in MODE 4.	6 hours 12 hours
D. One required SWIS DC electrical power subsystem battery connection resistance not within limit.	D.1	Restore the battery connection resistance to within the limit.	24 hours

Insert 3

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One required SWIS DC electrical power subsystem inoperable. OR Required Action and associated Completion Time of Condition D not met.	E.1	Declare the associated Service Water System train inoperable.	Immediately

Insert 23

Insert 24

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 127.8 V on float charge.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify post-to-post battery connection resistance of each cell-to-cell and terminal connection is ≤ 150 microhms for the Auxiliary Building batteries and ≤ 1500 microhms for the SWIS batteries.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters—Operating

LCO 3.8.7 The required Train A and Train B inverters shall be OPERABLE.

---NOTE---

Two inverters may be disconnected from their associated DC bus for ≤ 24 hours to perform an equalizing charge on their associated common battery, provided:

a. The associated AC vital buses are energized from their Class 1E constant voltage source transformers; and

b. All other AC vital buses are energized from their associated OPERABLE inverters.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required inverter inoperable.	A.1	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital bus de-energized. ----------------------------------- Restore inverter to OPERABLE status.	24 hours

Insert 25

Insert 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B.~~ C. Required Action and associated Completion Time of Condition A or B not met.	~~B~~C.1 Be in MODE 3.	6 hours
	AND	
	~~B~~C.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering MODE 4. ----- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to required AC vital buses.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more AC electrical power distribution subsystems inoperable for reasons other than Condition A, B, or C.	D.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours Insert 3
E. One or more AC vital buses inoperable.	E.1 Restore AC vital bus subsystem(s) to OPERABLE status.	8 hours Insert 3
F. One Auxiliary Building DC electrical power distribution subsystem inoperable.	F.1 Restore Auxiliary Building DC electrical power distribution subsystem to OPERABLE status.	2 hours Insert 3
Insert 26		
~~G~~ H. Required Action and associated Completion Time of Condition D, E, ~~or~~ F , or G not met.	~~G~~ H.1 Be in MODE 3. AND ~~G~~ H.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Be in MODE 4.	6 hours 12 hours
~~H~~ I. One Service Water Intake Structure (SWIS) DC electrical power distribution subsystem inoperable.	~~H~~ I.1 Declare the associated Service Water train inoperable.	Immediately
~~I. Two trains with inoperable distribution subsystems that result in a loss of safety function.~~	~~I.1 Enter LCO 3.0.3.~~	~~Immediately~~

5.5 Programs and Manuals

5.5.18 Control Room Envelope Habitability Program (continued)

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 SR 3.0.2 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.

5.5.19 Surveillance Frequency Control Program

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

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

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

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

Insert 27

INSERT 1

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable	A.1 Restore subsystem to OPERABLE status	7 days OR In accordance with the Risk Informed Completion Time Program.
B. ---------NOTES---------- 1. Not applicable when the second subsystem is deliberately made inoperable. 2. The following Section 5.5.20 constraints are applicable: b, c.2, c.3, d, e, f, g, and h. ------------------ Two subsystems inoperable	B.1 Restore one subsystem to OPERABLE status	1 hour OR In accordance with the Risk Informed Completion Time Program.
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3 AND C.2 Be in MODE 5	6 hours 36 hours

INSERT 1 (continued)

EXAMPLE 1.3-8

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

If a second subsystem is declared inoperable, Condition B may also be entered. The Condition is modified by two Notes. The first note states it is not applicable if the second subsystem is intentionally made inoperable. The second note provides restrictions applicable to these "loss of function" Conditions. The Required Actions of Condition B are not intended for voluntary removal of redundant subsystems from service. The Required Action is only applicable if one subsystem is inoperable for any reason and the second subsystem is found to be inoperable, or if both subsystems are found to be inoperable at the same time. If Condition B is applicable, at least one subsystem must be restored to OPERABLE status within 1 hour or Condition C must also be entered. The licensee may be able to apply a RICT or to extend the Completion Time beyond 1 hour, but not longer than 24 hours, if the requirements of the Risk Informed Completion Time Program are met. If two subsystems are inoperable and Condition B is not applicable (i.e., the second subsystem was intentionally made inoperable), LCO 3.0.3 is entered as there is no applicable Condition.

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

If the 7 day Completion Time clock of Condition A or the 1 hour Completion Time clock of Condition B have expired and subsequent changes in plant conditions result in exiting the applicability of the Risk Informed Completion Time Program without restoring the inoperable subsystem to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start.

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

INSERT 2

-------------------NOTES-------------------

1. Not applicable when a pressurizer safety valve is intentionally made inoperable.
2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.

INSERT 3

<u>OR</u>

In accordance with the Risk Informed Completion Time Program

INSERT 4

-------------------NOTES-------------------

1. Not applicable when the second block valve is intentionally made inoperable.
2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.

INSERT 5

C. -----------NOTES---------- 1. Not applicable when two or more ECCS accumulators are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two or more accumulators inoperable for reasons other than boron concentration not within limits.	C.1 Restore accumulators to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

INSERT 6

-------------------NOTES-------------------

1. Not applicable when the RWST is intentionally made inoperable.
2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.

INSERT 7

-------------------NOTES-------------------

1. Only applicable to penetration flow paths with two containment isolation valves.
2. Not applicable when the second containment isolation valve is intentionally made inoperable.
3. The following Section 5.5.20 constraints apply: parts b, c.2, c.3, d, e, f, g, and h.

INSERT 8

-------------------NOTES-------------------

1. Only applicable to penetration flow paths with only one containment isolation valve and a closed system.
2. Not applicable when the containment isolation valve is intentionally made inoperable.
3. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.

INSERT 9

B. ------------NOTES------------ 1. Not applicable when the second containment spray train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ----------------------------------- Two Containment Spray trains inoperable.	B. Restore one Containment Spray train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

INSERT 10

-------------------NOTES-------------------

1. Not applicable when three or more combinations of trains are intentionally made inoperable.
2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.

INSERT 11

H. Required Action and associated Completion Time of Condition G not met.	H.1 Be in MODE 3. <u>AND</u> H.2 Be in MODE 5.	6 hours 36 hours

INSERT 12

-------------------NOTES-------------------

1. Not applicable when second MSIV in a line is intentionally made inoperable.
2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.

INSERT 13

B. Two required ARV lines inoperable	B.1 Restore one ARV line to OPERABLE status	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
----------------NOTE------------------ 1. Not applicable when the third ARV line is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------------- C. Three required ARV lines inoperable	C.1 Restore one ARV line to OPERABLE status	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

INSERT 14

Intentionally Omitted

Note: During the final SNC internal review process for this License Amendment Request, Amendments 219 and 216 were approved for plant Farley. TSTF-412 amendments proposed a different Condition C which resulted in the deletion of this insert.

INSERT 15

-------------------NOTES-------------------

1. Not applicable when the CST is intentionally made inoperable.
2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g and h.

INSERT 16

B. -----------NOTE------------- 1. Not applicable when the second CCW train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ---------------------------------- Two CCW trains inoperable.	B.1 Restore one CCW train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program.

INSERT 17

B. -----------NOTE------------ 1. Not applicable when the second SWS train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ---------------------------------- Two SWS trains inoperable.	B.1 Restore one SWS train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program.

INSERT 18

-------------------NOTES-------------------

1. Not applicable when second CRACS train is intentionally made inoperable.
2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.

INSERT 19

E.1 Restore one CRACS train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

INSERT 20

F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 5.	6 hours 36 hours

INSERT 21

B. -----------NOTES------------- 1. Not applicable when the second ESF Room Cooler is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: Parts b, C.2, c.3, d, e, f, g, and h --- Two trains of the same ESF Room Cooler subsystem inoperable	B.1 Restore one of the same ESF Room Cooler subsystems to OPERABLE status	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

INSERT 22

H. ------------NOTES------------ 1. Not applicable when three or more AC sources are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ----------------------------------- Three or more required AC sources inoperable.	H.1 Restore required AC sources to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

INSERT 23

F. ------------NOTES------------ 1. Not applicable when a second DC power electrical subsystem is intentionally removed from service. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ----------------------------------- Two or more DC electrical subsystems inoperable that result in a loss of function	F.1 Restore required DC electrical subsystems to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

INSERT 24

G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. <u>AND</u> G.2 --------------NOTE--------------- LCO 3.0.4a is not applicable when entering MODE 4. --------------------------------------- Be in MODE 4.	6 hours 12 hours

INSERT 25

<table>
<tr>
<td>B. -----------NOTES--------------
1. Not applicable when the second required inverter is intentionally made inoperable.
2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.

Two or more required inverters inoperable.</td>
<td>A.1 Restore required inverters to operable status.</td>
<td>1 hour

<u>OR</u>

In accordance with the Risk Informed Completion Time Program</td>
</tr>
</table>

INSERT 26

G. ------------NOTE------------------ 1. Not applicable when two or more electrical power distribution trains are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. -------------------------------------- Two trains with inoperable electrical distribution subsystems that result in a loss of function.	G.1 Restore one train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program.

INSERT 27

5.5.20 Risk Informed Completion Time Program

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

a. The RICT may not exceed 30 days.
b. A RICT may only be utilized in MODE 1 and 2.
c. When a RICT is being used, any plant configuration change within the scope of the Configuration Risk Management 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. The RICT for these loss of function conditions may not exceed 24 hours.
f. 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 trains are considered “PRA Functional” as defined in Section 2.3.1 of NEI 06-09. However, the following additional constraints shall be applied to the criteria for “PRA Functional”.

    1. Any structures, systems, and components (SSC) credited in the PRA Functionality determination shall be the same SSCs relied upon to perform the specified Technical Specifications safety function.
    2. Design basis success criteria parameters shall be met for all design basis accident scenarios for establishing PRA Functionality, during a Technical Specifications loss of function condition, where a RICT is applied.

g. Upon entering a RICT for an emergent condition, the potential for a common cause (CC) failure must be addressed.

(continued)

If there is a high degree of confidence, based on the evidence collected, that there is no CC failure mechanism that could affect the redundant components, the RICT calculation may use nominal CC factor probability.

If a high degree of confidence cannot be established that there is no CC failure mechanism that could affect the redundant components, the RICT shall account for the increased possibility of CC failure. Accounting for the increased possibility of CC failure shall be accomplished by one of two methods. If one of the two methods listed below is not used, the Technical Specifications Front Stop shall not be exceeded.

1. The RICT calculation shall be adjusted to numerically account for the increased possibility of CC failure, in accordance with RG 1.177, as specified in Section A-1.3.2.1 of Appendix A of the RG. Specifically, when a component fails, the CC failure probability for the remaining components shall be increased to represent the conditional failure probability due to CC failure of these components, in order to account for the possibility the first failure was caused by a CC mechanism.

   OR

2. Prior to exceeding the front stop, RMAs not already credited in the RICT calculation shall be implemented. These RMAs shall target the success of the redundant and/or diverse SSC of the failed SSC and, if possible, reduce the frequency of initiating events which call upon the function(s) performed by the failed SSCs. Documentation of RMAs shall be available for NRC review.

h. A RICT entry is not permitted, or a RICT entry made shall be exited, for any condition involving a TS loss of function if a PRA Functionality determination that reflects the plant configuration concludes that the LCO cannot be restored without placing the TS inoperable trains in an alignment which results in a loss of functional level PRA success criteria.

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines

Attachment 3

Clean-Typed Technical Specifications Pages

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. ---------NOTES-------- 1. Not applicable when the second subsystem is deliberately made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. --------------------------- Two subsystems inoperable.	B.1 Restore one subsystem to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

EXAMPLES

<u>EXAMPLE 1.3-8</u> (continued)

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

If a second subsystem is declared inoperable, Condition B may also be entered. The Condition is modified by two Notes. The first note states it is not applicable if the second subsystem is intentionally made inoperable. The second note provides restrictions applicable to these “loss of function” Conditions. The Required Actions of Condition B are not intended for voluntary removal of redundant subsystems from service. The Required Action is only applicable if one subsystem is inoperable for any reason and the second subsystem is found to be inoperable, or if both subsystems are found to be inoperable at the same time. If Condition B is applicable, at least one subsystem must be restored to OPERABLE status within 1 hour or Condition C must also be entered. The licensee may be able to apply a RICT or to extend the Completion Time beyond 1 hour, but not longer than 24 hours, if the requirements of the Risk Informed Completion Time Program are met. If two subsystems are inoperable and Condition B is not applicable (i.e., the second subsystem was intentionally made inoperable), LCO 3.0.3 is entered as there is no applicable Condition.

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

If the 7 day Completion Time clock of Condition A or the 1 hour Completion Time clock of Condition B have expired and subsequent changes in plant conditions result in exiting the applicability of the Risk Informed Completion Time Program without restoring the inoperable subsystem to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-8 (continued)

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

IMMEDIATE COMPLETION TIME

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > the Low Temperature Overpressure Protection (LTOP) System applicability temperature specified in the PTLR.

---NOTE---
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTES---------- 1. Not applicable when a pressurizer safety valve is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4 with any RCS cold leg temperatures ≤ the LTOP System applicability temperature specified in the PTLR.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each PORV and each block valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve.	1 hour
	<u>AND</u>	
	B.2 Remove power from associated block valve.	1 hour
	<u>AND</u>	
	B.3 Restore PORV to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One block valve inoperable.	C.1	Place associated PORV in manual control.	1 hour
	AND		
	C.2	Restore block valve to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4.	12 hours
E. Two PORVs inoperable and not capable of being manually cycled.	E.1	Close associated block valves.	1 hour
	AND		
	E.2	Remove power from associated block valves.	1 hour
	AND		
	E.3	Be in MODE 3.	6 hours
	AND		
	E.4	Be in MODE 4.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. -----------NOTES---------- 1. Not applicable when the second block valve is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two block valves inoperable.	F.1 <u>AND</u> F.2	Place associated PORVs in manual control. Restore one block valve to OPERABLE status.	1 hour 2 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
G. Required Action and associated Completion Time of Condition F not met.	G.1 <u>AND</u> G.2	Be in MODE 3. Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-------------------------------NOTES----------------------------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Three ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

--NOTE--
In MODE 3, with RCS pressure > 1000 psig, the accumulators may be inoperable for up to 12 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----------NOTES---------- 1. Not applicable when two or more ECCS accumulators are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two or more accumulators inoperable for reasons other than boron concentration not within limits.	C.1 Restore accumulators to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Reduce RCS pressure to ≤ 1000 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7555 gallons (31.4%) and ≤ 7780 gallons (58.4%).	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 601 psig and ≤ 649 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2200 ppm and ≤ 2500 ppm.	In accordance with the Surveillance Frequency Control Program <u>AND</u> -------NOTE------- Only required to be performed for affected accumulators --------------------- Once within 6 hours after each solution volume increase of ≥ 12% level, indicated, that is not the result of addition from the refueling water storage tank
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is ≥ 2000 psig.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS — Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

--NOTES--

1. In MODE 3, the Residual Heat Removal or the Centrifugal Charging Pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

2. Upon entry into MODE 3 from MODE 4, the breaker or disconnect device to the valve operators for MOVs 8706A and 8706B may be locked open for up to 4 hours to allow for repositioning from MODE 4 requirements.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more trains inoperable.	A.1	Restore train(s) to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	C.1	Enter LCO 3.0.3.	Immediately

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. -----------NOTES---------- 1. Not applicable when the RWST is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. AND C.2 Verify a door is closed in the affected air lock. AND C.3 Restore air lock to OPERABLE status.	Immediately 1 hour 24 hours OR In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

ACTIONS

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -----------NOTES---------- 1. Only applicable to penetration flow paths with two containment isolation valves. 2. Not applicable when the second Containment isolation valve is intentionally made inoperable. 3. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ One or more penetration flow paths with two containment isolation valves inoperable except for purge valve penetration leakage not within limit.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ------------NOTE------------ 1. Only applicable to penetration flow paths with only one containment isolation valve and a closed system. 2. Not applicable when the containment isolation valve is intentionally made inoperable. 3. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. -------------------------------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours OR In accordance with the Risk Informed Completion Time Program
	AND		
	C.2	-------------NOTE------------ Isolation devices in high radiation areas may be verified by use of administrative means. --------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. -----------NOTES---------- 1. Not applicable when the second containment spray train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two Containment Spray trains inoperable.	B. Restore one Containment Spray train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	------------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Be in MODE 4.	54 hours
D. One containment cooling train inoperable.	D.1	Restore containment cooling train to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
E. Two containment cooling trains inoperable.	E.1	Restore one containment cooling train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
F Required Action and associated Completion Time of Condition D or E not met.	F.1	Be in MODE 3.	6 hours
	AND		
	F.2	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------- Be in MODE 4.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. -----------NOTES---------- 1. Not applicable when three or more combinations of trains are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Any combination of three or more trains inoperable.	G.1 Restore required trains to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
H. Required Action and associated Completion Time of Condition G not met.	H.1 Be in MODE 3. <u>AND</u> H.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	----------------------------NOTE-------------------------------- Not required to be met for system vent flow paths opened under administrative control. --- Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.6.2	Operate each required containment cooling train fan unit for $\geq$ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.3	Verify each containment cooling train cooling water flow rate is $\geq$ 1600 gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.7	Verify each containment cooling train starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.8	Verify each spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs per steam line shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when one MSIV in each steam line is closed.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each steam line.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam lines with one MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. -----------NOTES--------- 1. Not applicable when second MSIV in a line is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ One or more steam lines with two MSIVs inoperable in MODE 1.	B.1 Restore one MSIV to OPERABLE status in affected steam line.	4 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours
D. One or more steam lines with one MSIV inoperable in MODE 2 or 3.	D.1 Verify one MSIV closed in affected steam line.	7 days AND Once per 7 days thereafter
E. One or more steam lines with two MSIVs inoperable in MODE 2 or 3.	E.1 Verify one MSIV closed in affected steam line.	4 hours AND Once per 7 days thereafter
F. Required Action and associated Completion Time of Condition D or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------------------NOTE------------------------------- Only required to be performed in MODES 1 and 2. --- Verify closure time of each MSIV is ≤ 7 seconds.	In accordance with the INSERVICE TESTING PROGRAM

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Three ARV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ARV line inoperable.	A.1 Restore required ARV line to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Two required ARV lines inoperable.	B.1 Restore one ARV line to OPERABLE status.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
C. -----------NOTES---------- 1. Not applicable when the third ARV line is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Three required ARV lines inoperable.	C.1 Restore one ARV line to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ARV.	In accordance with the Surveillance Frequency Control Program
SR 3.7.4.2	Verify one complete cycle of at least one manual isolation valve in each ARV Line.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Turbine driven AFW train inoperable due to one inoperable steam supply. <u>OR</u> ------------NOTE----------- Only applicable if MODE 2 has not been entered following refueling. ------------------------------- One turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One AFW train inoperable for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Turbine driven AFW train inoperable due to one inoperable steam supply. AND One motor driven AFW train inoperable.	C.1 Restore the steam supply to the turbine driven train to OPERABLE status. OR C.2 Restore the motor driven AFW train to OPERABLE status.	24 hours 24 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met. OR Two AFW trains inoperable for reasons other than Condition C.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours
E. Three AFW trains inoperable.	E.1 --------------NOTE--------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ------------------------------------- Initiate action to restore one AFW train to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	------------------------------NOTE------------------------------ AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. --- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	----------------------------NOTE-------------------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 1005 psig in the steam generator. --- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM.
SR 3.7.5.3	----------------------------NOTE-------------------------------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. --- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.4	---------------------------NOTES--------------------------- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 1005 psig in the steam generator. 2. AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. --- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.5	Verify the turbine driven AFW pump steam admission valves open when air is supplied from their respective air accumulators.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTES---------- 1. Not applicable when the CST is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ CST inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply. <u>AND</u> A.2 Restore CST to OPERABLE status.	4 hours <u>AND</u> Once per 12 hours thereafter 7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program.
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST level is ≥ 164,000 gal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by CCW. ---------------------------------- Restore CCW train to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program.
B. -----------NOTES---------- 1. Not applicable when the second CCW train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two CCW trains inoperable.	B.1	Restore one CCW train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------	
		Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

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

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS train inoperable.	A.1 -----------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources — Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by SWS. --------------------------------- Restore SWS train to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -----------NOTES---------- 1. Not applicable when the second SWS train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two SWS trains inoperable.	B.1	Restore one SWS train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
C. One SWS automatic turbine building isolation valve inoperable in each SWS train.	C.1	Restore both inoperable turbine building isolation valves to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. --------------------------------- Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1	------------------------------NOTE------------------------------ Isolation of SWS flow to individual components does not render the SWS inoperable. -- Verify each accessible SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each SWS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.4	Verify the integrity of the SWS buried piping by visual inspection of the ground area.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two CRACS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	D.1 Suspend CORE ALTERATIONS. AND D.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
E. -----------NOTES---------- 1. Not applicable when the second CRACS train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two CRACS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Restore one CRACS train to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CRACS train has the capability to remove the assumed heat load.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.19 Engineered Safety Feature (ESF) Room Coolers

LCO 3.7.19 ESF Room Coolers shall be OPERABLE.

APPLICABILITY: When associated ESF equipment is required to be OPERABLE.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each ESF Room Cooler subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ESF Room Cooler subsystem Train inoperable.	A.1 Restore the affected ESF Room Cooler subsystem Train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. -----------NOTES---------- 1. Not applicable when the second ESF Room Cooler is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two trains of the same ESF Room Cooler subsystem inoperable.	B.1 Restore one of the same ESF Room Cooler subsystems to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.19.1	Verify each ESF Room Cooler system manual valve servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.19.2	Verify each ESF Room Cooler fan starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore required offsite circuit to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One DG set inoperable.	------------------NOTE------------------ LCO 3.0.4c is applicable when only one of the three DGs is inoperable. --		
	B.1	Perform SR 3.8.1.1 for the required offsite circuit(s).	2 hours <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
	B.2	Declare required feature(s) supported by the inoperable DG set inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	B.3.1	Determine OPERABLE DG set is not inoperable due to common cause failure.	24 hours
	<u>OR</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.2	Perform SR 3.8.1.6 for OPERABLE DG set.	24 hours
	AND		
	B.4	Restore DG set to OPERABLE status.	10 days OR In accordance with the Risk Informed Completion Time Program
C. Two required offsite circuits inoperable.	C.1	Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required features
	AND		
	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable. <u>AND</u> One DG set inoperable.	----------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems — Operating," when Condition D is entered with no AC power source to any train. --	
	D.1 Restore required offsite circuit to OPERABLE status.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>OR</u>	
	D.2 Restore DG set to OPERABLE status.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two DG sets inoperable.	E.1	Restore one DG set to OPERABLE status.	8 hours if DG 1-2A and DG 1(2)B are inoperable <u>OR</u> In accordance with the Risk Informed Completion Time Program <u>OR</u> 24 hours if DG 1C and DG 1(2)B are inoperable <u>OR</u> In accordance with the Risk Informed Completion Time Program
F. Required Action and associated Completion Time of Condition C or E not met.	F.1	Be in MODE 3.	6 hours
G. One automatic load sequencer inoperable.	G.1	Restore automatic load sequencer to OPERABLE status.	12 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. -----------NOTES---------- 1. Not applicable when three or more AC sources are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Three or more required AC sources inoperable.	H.1	Restore required AC sources to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program, not to exceed 72 hours
I. Required Action and associated Completion Time of Condition A, B, D, G, or H not met.	I.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	I.2	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	----------------------------NOTES------------------------------ 1. Performance of SR 3.8.1.6 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.6 must be met. --- Verify each DG starts from standby conditions and achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-------------------------------NOTES---------------------------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.6. --- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2700 kW and ≤ 2850 kW for the 2850 kW DG and ≥ 3875 kW and ≤ 4075 kW for the 4075 kW DGs.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify each day tank contains ≥ 900 gal of fuel oil for the 4075 kW DGs and 700 gal of fuel oil for the 2850 kW DG.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Verify the fuel oil transfer system operates to transfer fuel oil from storage tank to the day tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.6	------------------------------NOTE------------------------------- All DG starts may be preceded by an engine prelube period. --- Verify each DG starts from standby condition and achieves in ≤ 12 seconds, voltage ≥ 3952 V and frequency ≥ 60 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.7	-----------------------------NOTE------------------------------- This Surveillance shall not normally be performed in MODE 1 or 2. However, this surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. --	
	Verify manual transfer of AC power sources from the normal offsite circuit to the alternate required offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.8	Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the speed is ≤ 75% of the difference between nominal speed and the overspeed trip setpoint; and b. Following load rejection, the voltage is ≥ 3740 V and ≤ 4580 V.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	-------------------------------NOTES---------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. ---	
	Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 12 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage ≥ 3740 V and ≤ 4580 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	-----------------------------NOTE----------------------------- All DG starts may be preceded by prelube period. --- Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and: a. In ≤ 12 seconds after auto-start and during tests, achieves voltage ≥ 3952 V; b. In ≤ 12 seconds after auto-start and during tests, achieves frequency ≥ 60 Hz; c. Operates for ≥ 5 minutes and maintains a steady state generator voltage and frequency of ≥ 3740 V and ≤ 4580 V and ≥ 58.8 Hz and ≤ 61.2 Hz; -----------------------------NOTE----------------------------- SR 3.8.1.10.d and e shall not be performed in MODE 1 or 2. --- d. Permanently connected loads remain energized from the offsite power system; and e. Emergency loads are energized from the offsite power system.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.11	Verify each DG's automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus and/or an actual or simulated ESF actuation signal except: a. Engine overspeed; b. Generator differential current; and c. Low lube oil pressure.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.12	-------------------------------NOTE------------------------------- Momentary transients below the minimum load specified do not invalidate this test. ---	
	Verify each DG operates for ≥ 24 hours: a. For ≥ 2 hours loaded ≥ 4353 for the 4075 kW DGs and ≥ 3100 kW for the 2850 kW DG; and b. For the remaining hours of the test loaded ≥ 4075 kW for the 4075 kW DGs and ≥ 2850 kW for the 2850 kW DG.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.13	-------------------------------NOTES----------------------------- 1. This Surveillance shall be performed within 10 minutes of shutting down the DG after the DG has operated ≥ 2 hours loaded ≥ 4075 kW for the 4075 kW DGs and ≥ 2850 kW for the 2850 kW DG. Momentary transients below the minimum load specified do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. ---	
	Verify each DG starts and achieves, in ≤ 12 seconds, voltage ≥ 3952 V and frequency ≥ 60 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.14	-------------------------------NOTE------------------------------- This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, this surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. ---	
	Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by returning DG to ready-to-load operation.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.16	Verify interval between each sequenced load block is within ± 10% of design interval or 0.5 seconds, whichever is greater, for each emergency load sequencer.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.17	------------------------------NOTES---------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. --- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 12 seconds,	In accordance with the Surveillance Frequency Control Program (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.17 (continued)	2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage $\geq$ 3740 V and $\leq$ 4580 V, 4. achieves steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for $\geq$ 5 minutes.	
SR 3.8.1.18	-------------------------------NOTE------------------------------ Testing of the shared Emergency Diesel Generator (EDG) set (EDG 1-2A or EDG 1C) on either unit may be used to satisfy this surveillance requirement for these EDGs for both units. --- Verify each DG does not trip and voltage is maintained $\leq$ 4990 V and $\geq$ 3330 V during and following a load rejection of $\geq$ 1200 kW and $\leq$ 2400 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.19	-------------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period. --- Verify when started simultaneously from standby condition, each DG achieves, in $\leq$ 12 seconds, voltage $\geq$ 3952 V and frequency $\geq$ 60 Hz.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources — Operating

LCO 3.8.4 The Train A and Train B Auxiliary Building and Service Water Intake Structure (SWIS) DC electrical power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One Auxiliary Building DC electrical power subsystem inoperable.	A.1	Restore the Auxiliary Building DC electrical power subsystem to OPERABLE status.	2 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One Auxiliary Building DC electrical power subsystem with battery connection resistance not within limit.	B.1	Restore the battery connection resistance to within limit.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------------------------------- Be in MODE 4.	12 hours
D. One required SWIS DC electrical power subsystem battery connection resistance not within limit.	D.1	Restore the battery connection resistance to within the limit.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
E. One required SWIS DC electrical power subsystem inoperable. <u>OR</u> Required Action and associated Completion Time of Condition D not met.	E.1	Declare the associated Service Water System train inoperable.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. ------------NOTES----------- 1. Not applicable when a second DC power electrical subsystem is intentionally removed from service. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two or more DC electrical subsystems inoperable that result in a loss of function.	F.1 Restore required DC electrical subsystems to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. <u>AND</u> G.2 -------------NOTE--------------- LCO 3.0.4a is not applicable when entering MODE 4. ------------------------------------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 127.8 V on float charge.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. <u>OR</u> Verify post-to-post battery connection resistance of each cell-to-cell and terminal connection is ≤ 150 microhms for the Auxiliary Building batteries and ≤ 1500 microhms for the SWIS batteries.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.4	Remove visible terminal corrosion, verify battery cell-to-cell and terminal connections are coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.5	Verify post-to-post battery connection resistance of each cell-to-cell and terminal connection is ≤ 150 microhms for the Auxiliary Building batteries and ≤ 1500 microhms for the SWIS batteries	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.6	------------------------------NOTE------------------------------ This Surveillance may be performed in MODE 1, 2, 3, 4, 5, or 6 provided spare or redundant charger(s) placed in service are within surveillance frequency to maintain DC subsystem(s) OPERABLE. --- Verify each required Auxiliary Building battery charger supplies ≥ 536 amps at ≥ 125 V for ≥ 4 hours and each required SWIS battery charger supplies ≥ 3 amps at ≥ 125 V for ≥ 4 hours.	 In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.7	----------------------------NOTES------------------------------ 1. The performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 once per 60 months. 2. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test at any time. 3. This Surveillance shall not normally be performed for the Auxiliary Building batteries in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. -- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design load profile described in the Final safety Analysis Report, Section 8.3.2, by subjecting the battery to a service test.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.8	-------------------------------NOTE------------------------------- This Surveillance shall not normally be performed for the Auxiliary Building batteries in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. ---	
	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	In accordance with the Surveillance Frequency Control Program <u>AND</u> 18 months when battery shows degradation or has reached 85% of expected life or 17 years, whichever comes first

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters — Operating

LCO 3.8.7 The required Train A and Train B inverters shall be OPERABLE.

---NOTE--
Two inverters may be disconnected from their associated DC bus for ≤ 24 hours to perform an equalizing charge on their associated common battery, provided:

a. The associated AC vital buses are energized from their Class 1E constant voltage source transformers; and

b. All other AC vital buses are energized from their associated OPERABLE inverters.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 ------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital bus de-energized. ------------------------------- Restore inverter to OPERABLE status.	 24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. -----------NOTES---------- 1. Not applicable when the second required inverter is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two or more required inverters inoperable.	B.1 Restore required inverters to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 ------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to required AC vital buses.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more AC electrical power distribution subsystems inoperable for reasons other than Condition A, B, or C.	D.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
E. One or more AC vital buses inoperable.	E.1 Restore AC vital bus subsystem(s) to OPERABLE status.	8 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
F. One Auxiliary Building DC electrical power distribution subsystem inoperable.	F.1 Restore Auxiliary Building DC electrical power distribution subsystem to OPERABLE status.	2 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. -----------NOTES---------- 1. Not applicable when two or more electrical power distribution trains are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h. ------------------------------ Two trains with inoperable electrical distribution subsystems that result in a loss of function.	G.1 Restore one train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
H. Required Action and associated Completion Time of Condition D, E, F, or G not met.	H.1 Be in MODE 3. <u>AND</u> H.2 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Be in MODE 4.	6 hours 12 hours
I. One Service Water Intake Structure (SWIS) DC electrical power distribution subsystem inoperable.	I.1 Declare the associated Service Water train inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

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

5.5.20 Risk Informed Completion Time Program

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

a. The RICT may not exceed 30 days.
b. A RICT may only be utilized in MODE 1 and 2.
c. When a RICT is being used, any plant configuration change within the scope of the Configuration Risk Management 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. The RICT for these loss of function conditions may not exceed 24 hours.
f. 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 trains are considered "PRA Functional" as defined in Section 2.3.1 of NEI 06-09. However, the following additional constraints shall be applied to the criteria for "PRA Functional".

    1. Any structures, systems, and components (SSCs) credited in the PRA Functionality determination shall be the same SSCs relied upon to perform the specified Technical Specifications safety function.
    2. Design basis success criteria parameters shall be met for all design basis accident scenarios for establishing PRA Functionality, during a Technical Specifications loss of function condition, where a RICT is applied.

g. Upon entering a RICT for an emergent condition, the potential for a common cause (CC) failure must be addressed.

(continued)

5.5.20 Risk Informed Completion Time Program (continued)

If there is a high degree of confidence, based on the evidence collected, that there is no CC failure mechanism that could affect the redundant components, the RICT calculation may use nominal CC factor probability.

If a high degree of confidence cannot be established that there is no CC failure mechanism that could affect the redundant components, the RICT shall account for the increased possibility of CC failure. Accounting for the increased possibility of CC failure shall be accomplished by one of two methods. If one of the two methods listed below is not used, the Technical Specifications Front Stop shall not be exceeded.

1. The RICT calculation shall be adjusted to numerically account for the increased possibility of CC failure, in accordance with RG 1.177, as specified in Section A-1.3.2.1 of Appendix A of the RG. Specifically, when a component fails, the CC failure probability for the remaining components shall be increased to represent the conditional failure probability due to CC failure of these components, in order to account for the possibility the first failure was caused by a CC mechanism.

   OR

2. Prior to exceeding the front stop, RMAs not already credited in the RICT calculation shall be implemented. These RMAs shall target the success of the redundant and/or diverse SSCs of the failed SSC and, if possible, reduce the frequency of initiating events which call upon the function(s) performed by the failed SSCs. Documentation of RMAs shall be available for NRC review.

h. A RICT entry is not permitted, or a RICT entry made shall be exited, for any condition involving a TS loss of function if a PRA Functionality determination that reflects the plant configuration concludes that the LCO cannot be restored without placing the TS inoperable trains in an alignment which results in a loss of functional level PRA success criteria.

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, “Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines”

Attachment 4

Marked-Up Technical Specifications Bases Pages (Information Only)

BASES

ACTIONS

A.1

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

Bases Insert 1

Bases Insert 2

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ the LTOP System applicability temperature specified in the PTLR within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below the LTOP System applicability temperature specified in the PTLR, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Pressurizer safety valves are to be tested in accordance with the requirements of the ASME OM Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 1% for OPERABILITY.

BASES

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. FSAR, Chapter 5.2, 5.5, 15.2, 15.3 and 15.4.
3. WCAP-7769, Rev. 1, June 1972.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).
5. (Add SE reference here.)

BASES

ACTIONS (continued)

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

If one PORV is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

Bases Insert 1

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs are not capable of mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

Bases Insert 1

(continued)

Information Only

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4, 5, and 6, the PORVs are not required OPERABLE.

E.1, E.2, E.3, and E.4

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two PORVs inoperable. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4, 5, and 6, the PORVs are not required OPERABLE.

Bases Insert 3

F.1 and F.2

If two block valves are inoperable, it is necessary to restore at least one block valve within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

Bases Insert 4

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV in MODE 3 or 4. The PORVs are stroke tested during MODES 3 or 4 with the associated block valves closed in order to limit the uncertainty introduced by testing the PORVs at lesser system temperatures than expected during actual operating conditions. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature conditions, prior to entering MODE 1 or 2.

SR 3.4.11.3

SR 3.4.11.3 requires a complete cycle of each PORV using the backup PORV control system. This surveillance verifies the capability to operate the PORVs using the backup nitrogen supply system. Additionally, this surveillance ensures the correct function of the associated nitrogen supply system valves. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. FSAR Sections 5.5 and 15.2.
3. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," June 25, 1990.
4. (Add SE reference here.)

BASES

ACTIONS (continued)

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of two 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 24 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. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified in WCAP-15049-A, Rev. 1 (Ref. 3).

Bases Insert 5

D

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

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~D.1~~

~~If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.~~

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

Information Only

BASES

SURVEILLANCE REQUIREMENTS	SR 3.5.1.5 (continued) Should closure of a valve occur below 2000 psig, the SI signal provided to the valves would open a closed valve in the event of a LOCA.
REFERENCES	1. FSAR, Chapter 15. 2. 10 CFR 50.46 3. WCAP-15049-A, Rev. 1, April 1999. 4. NUREG-1366, February 1990. 5. (Add SE reference.)

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

Bases Insert 1

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

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

BASES

REFERENCES (continued)

3. FSAR, Section 6, "Engineered Safety Features."
4. FSAR, Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.
7. ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).
8. (Add SE reference here.)

ACTIONS (continued)

These administrative controls consist of (1) Stroking valve Q1(2)G31V010 open and then closed prior to circulating the RWST water through the Spent Fuel Pool Purification System (2) establishing a designated operator to control the valve and (3) establishing a preplanned communication method between the operator and Shift Supervisor. In this way, the flow path can be rapidly isolated in the event of a Reactor Trip or at the direction of the Shift Supervisor. These Notes are to allow recirculation and sampling of the RWST through the Spent Fuel Pool Purification System for filtering as well as operation of the reverse osmosis system to remove silica. These Notes can only be applied during the next two fuel Cycles for each Unit. These Notes cannot be used after Refueling Outages 1R26 (Spring 2015) and 2R24 (Spring 2016).

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

Bases Insert 6

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

Bases Insert 7

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Remaining within the applicability of the LCO is

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

The RWST water volume should be verified to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.4.3

The boron concentration of the RWST should be verified to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.

2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

3. (Add reference to SE here.)

ACTIONS

C.1, C.2, and C.3 (continued)

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

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

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

Bases Insert 1

D.1 and D.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage

(continued)

Information Only

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Section 6.2.
3. NEL Letter NEL-02-0144, dated June 25, 2002.
4. (Add reference to SE here.)

Information Only

BASES

ACTIONS

A.1 and A.2 (continued)

active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with forward flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

Bases Insert 1

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1~~.~~ Bases Insert 1 In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Bases Insert 8

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

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a

(continued)

Information Only

BASES

ACTIONS

C.1 and C.2 (continued)

single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time~~. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4~~. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Bases Insert 1

Condition C is modified by ~~a Note indicating~~ three notes. The first Note indicates that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Ref. 5. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system. FSAR Table 6.2-31 identifies the following containment isolation valves as being in a Type III penetration (closed system) and having only one containment isolation valve: Q1/2 B13V026B (Pressurizer pressure generator).

Bases Insert 8a

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

(continued)

BASES

REFERENCES

1. FSAR, Section 15.
2. FSAR, Section 6.2.
3. Not used.
4. Not used.
5. Standard Review Plan 6.2.4.
6. (Add SE reference here.)

Information Only

BASES

APPLICABILITY

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

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

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and low probability of a DBA occurring during this period. Bases Insert 1

Bases Insert 9

C

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

or trains

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 54 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 7). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth.

(continued)

Information Only

BASES

ACTIONS

C

~~B~~.1 and ~~B~~.2 (continued)

As stated in Reference 7, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action ~~B~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 4 allows 48 hours to restore the containment spray train to OPERABLE status in MODE 3. This is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

D

~~C~~.1

Bases Insert 1

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

(continued)

Information Only

BASES

ACTIONS

F   Bases Insert 1

E   ~~D~~.1

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

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

D

If the Required Action and associated Completion Time of Condition ~~C~~ or ~~D~~ of this LCO are not met, the plant must be brought to a MODE in which overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 7). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 7, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

(continued)

Information Only

F

BASES

ACTIONS

~~E~~.1 and ~~E~~.2 (continued)

Required Action ~~E~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

G

~~F~~.1

With ~~two containment spray trains or~~ any combination of three or more containment spray and cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, ~~LCO 3.0.3 must be entered immediately.~~

Bases Insert 10

Bases Insert 11

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

Information Only

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.9 (continued)

required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. 10 CFR 50, Appendix K.
3. FSAR, Section 6.2.
4. FSAR, Section 7.3.
5. FSAR, Section 15.
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.
7. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.
8. (Add SE reference here.)

BASES

LCO (continued)

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when one MSIV in each steam line is closed, when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally most of the MSIVs are closed, and the steam generator energy is low.

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

ACTIONS

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each steam line. The Completion Time(s) of the inoperable MSIV Systems will be tracked separately for each steam line starting from the time the Condition was entered for that steam line.

A.1

With one MSIV inoperable in one or more steam lines in MODE 1, action must be taken to restore the inoperable MSIV to OPERABLE status within 72 hours. Some repairs to the MSIV can be made with the unit at power. The 72 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time that would require the MSIVs to close and the remaining OPERABLE MSIV in the steam line. This Completion Time is also consistent with the Completion Times provided for a single inoperable train in other ESF systems that contain redundant trains of equipment.

Bases Insert 1

(continued)

Information Only

BASES

ACTIONS (continued)

B.1

With two MSIVs inoperable in one or more steam lines in MODE 1, action must be taken to restore one MSIV to OPERABLE status in the affected steam line(s) within 4 hours. In this Condition, the affected steam line has no OPERABLE automatic isolation capability. The 4-hour Completion Time allows for minor repairs or trouble shooting that may prevent a unit shutdown to MODE 2 and is reasonable considering the low probability of an accident occurring during this time that would require the MSIVs to close and the reduced potential for a plant transient (shutdown to MODE 2) provided by the 4 hours allowed for restoration.

Bases Insert 1

Bases Insert 12

C.1

If the MSIV cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a Mode in which the ACTIONS provide the option to close the inoperable MSIV and accomplish the required safety function by isolating the affected steam line. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition D or E entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner without challenging unit systems.

D.1

Required Action D.1 is applicable when one or more steam lines have a single inoperable MSIV in MODE 2 or 3. Since the MSIVs are required OPERABLE in MODES 2 and 3, the inoperable MSIV(s) may either be restored to OPERABLE status or the affected steam line isolated by closing at least one MSIV in that steam line. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 7 day Completion Time is reasonable considering the plant condition, the low probability of an event occurring that would require the MSIV to close, and the remaining OPERABLE redundant MSIV in the affected steam line(s).

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, and the affected steam line is isolated by a closed MSIV, the MSIV must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

accident and containment analyses. This Surveillance is normally performed while returning the unit to operation following a refueling outage.

The Frequency is in accordance with the Inservice Testing Program, which encompasses the ASME OM Code (Ref. 5). Operating experience has shown that these components usually pass the Surveillance when performed in accordance with the Inservice Testing Program. Therefore, the Frequency is acceptable from a reliability standpoint.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. If desired, this allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated. This surveillance may be performed in lower modes but must be performed prior to entry into MODE 2.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 6.2.
3. FSAR, Section 15.4.2.
4. 10 CFR 100.11.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).
6. (Add SE reference here.)

Information Only

BASES

LCO (continued)

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Dump System.

An ARV is considered OPERABLE (even if isolated) when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand, either remotely or locally via manual control.

APPLICABILITY

In MODES 1, 2, and 3, the ARVs are required to be OPERABLE.

In MODE 4, the pressure and temperature limitations are such that the probability of an SGTR event requiring ARV operation is low. In addition, the RHR system is available to provide the decay heat removal function in MODE 4. Therefore, the ARVs are not required to be OPERABLE in MODE 4 to satisfy the safety analysis assumptions of the DBA. However, the capability to remove decay heat from a SG required to be OPERABLE in MODE 4 by LCO 3.4.6, "RCS Loops – MODE 4" is implicit in the requirement for an OPERABLE SG and may require the associated ARV be capable of removing that heat if the normal decay heat removal system (steam dump) is not available.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ARV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ARV lines, a nonsafety grade backup in the Steam Dump System, and MSSVs.

Bases Insert 1

~~B.1~~

~~With two or more ARV lines inoperable, action must be taken to restore all but one ARV line to OPERABLE status. Since the manual isolation valves can be closed to isolate an ARV, some repairs may~~

Information Only

ACTIONS

~~B.1 (continued)~~

~~be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ARV lines, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the ARV lines.~~

Bases Insert 13

D

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

If the ARV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ARVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the ARVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ARV during a unit cooldown may satisfy this requirement. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.4.2

The function of the manual isolation valve is to isolate a failed open ARV. Cycling the manual isolation valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the manual isolation valve during unit cooldown may satisfy this requirement. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 15.4.3.
3. (Add reference to SE here.)

BASES

ACTIONS

A.1 (continued)

a. For the inoperability of a steam supply to the turbine driven AFW pump, the 7 day Completion time is reasonable since there is a redundant steam supply line for the turbine driven pump.

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling, the 7 day Completion time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps; and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Bases Insert 1

Condition A is modified by a Note which limits the applicability of the Condition to when the unit has not entered MODE 2 following a refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

With one of the required AFW trains (pump or flow path) inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. A flow path is inoperable if it is blocked such that the required AFW flow cannot be delivered. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

Bases Insert 1

(continued)

BASES

SURVEILLANCE

SR 3.7.5.4 (continued)

The second Note states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

SR 3.7.5.5

This SR verifies that the air stored in turbine-driven AFW pump steam admission valve air accumulators is sufficient to open valves Q1(2)N12V001A-A and Q1(2)N12V001B-B. Each steam admission valve has an air accumulator associated with it. The air accumulators provide sufficient air to ensure the operation of the steam admission valves for turbine-driven AFW pump during a loss of power or other failure of the normal air supply. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.5.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).
3. (Add SE reference here.)

APPLICABILITY

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

In MODE 4, 5, or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup supply (Service Water System) should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the Service Water supply to the AFW pumps are OPERABLE, and that the Service Water System is capable of supplying water to the AFW pumps. The CST must be restored to OPERABLE status within 7 days, because the Service Water System does not supply the preferred quality of SG feedwater and may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.

Bases Insert 1

Bases Insert 15

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.2.6.
2. FSAR, Chapter 6.
3. FSAR, Chapter 15.
4. AFW – FSD A-181010.
5. CALC. BM 95-0961-001, Rev. 5, Verification of CST Sizing Basis.
6. (Add reference to SE here.)

Information Only

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," be entered if an inoperable CCW train results in an inoperable RHR loop. This note is only applicable in MODE 4. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

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

Bases Insert 1

Bases Insert 16

C

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

If the CCW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 2, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

(continued)

Information Only

C

BASES

ACTIONS

B.1 and B.2 (continued)

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW System. The Note is applicable to CCW loads and does not include components required for CCW OPERABILITY.

Verifying the correct alignment for accessible manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. The accessibility of the CCW valves is evaluated on a case by case basis considering such things as ALARA concerns and personnel safety as well as valve enclosures or barricades blocking access to the valves. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

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

(continued)

Information Only

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated Safety Injection actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.2.2.

2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

3. (Add SE reference here.)

Information Only

BASES

ACTIONS

A.1

If one SWS train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SWS train could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," should be entered if an inoperable SWS train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," should be entered if an inoperable SWS train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

Bases Insert 1

Bases Insert 17

C

~~B~~.1

With one automatic turbine building isolation valve inoperable in each SWS train, the inoperable valves must be restored to OPERABLE status within 72 hours. With the unit in this condition, the remaining OPERABLE SWS turbine building isolation valves in each train are adequate to perform the SWS non-essential load isolation function; however, the overall reliability of the function is reduced. The 72 hour Completion Time is based on the fact that the remaining OPERABLE automatic turbine building isolation valves in each SWS train ensure the SWS trains remain fully capable of performing the required safety function and the low probability of an event occurring during this time period that would require the isolation function of these valves.

D

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

If the SWS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

(continued)

D

BASES

ACTIONS

~~C~~.1 and ~~C~~.2 (continued)

Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 4, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action ~~C~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the SWS. The Note is applicable to SWS loads and does not include components required for SWS OPERABILITY.

Verifying the correct alignment for accessible manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. The accessibility of the SWS valves is evaluated on a case by case basis

(continued)

Information Only

BASES

REFERENCES

1. FSAR, Section 9.2.1.
2. FSAR, Section 6.2.
3. FSAR, Section 5.1.
4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.
5. (Add reference to SE here.)

Information Only

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1 and D.2

During movement of irradiated fuel assemblies, or during CORE ALTERATIONS, with two CRACS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

Bases Insert 18

If both CRACS trains are inoperable in MODE 1, 2, 3, or 4, the control room CRACS may not be capable of performing its intended function. Therefore, ~~LCO 3.0.3 must be entered immediately.~~

Bases Insert 19

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the safety analyses in the control room. This SR consists of system testing. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.4.
2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.
3. (Add reference to SE here.)

Information Only

BASES

ACTIONS (continued)

C

<u>A.1</u>

If one train of a required ESF Room Cooler subsystem is inoperable, action must be taken to restore the subsystem train to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ESF Room Cooler subsystem train is adequate to perform the heat removal function for its associated ESF equipment.

Bases Insert 1

Bases Insert 21

<u>~~B~~.1 and ~~B~~.2</u>

If the ESF Room Cooler subsystem train cannot be restored to OPERABLE status within the associated Completion Time ~~or two trains of the same ESF Room Cooler subsystem are inoperable~~, the unit must be placed in a MODE in which overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and MODE 4 within 12 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 2, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

C

Required Action ~~B~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

Information Only

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.19.1

Verifying the correct alignment for manual valves servicing safety-related equipment provides assurance that the proper flow paths exist for ESF Room Cooler operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

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

SR 3.7.19.2

This SR verifies proper operation of the ESF Room Cooler fans on an actual or simulated actuation signal. Depending on the room cooler, this may be manual, high room temperature, an equipment running signal, or some combination. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Each Room Cooler Fan can be placed in Run mode locally. With the Room Cooler in the Run mode, all automatic functions are being met and the Room Cooler is considered OPERABLE.

REFERENCES

1. FSAR, Section 9.4.

2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

3. (Add SE reference here.)

BASES

ACTIONS

A.3 (continued)

this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

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

Bases Insert 1

B.1

The Condition B Required Actions are modified by a Note that is applicable when only one of the three individual DGs is inoperable. The note permits the use of the provisions of LCO 3.0.4c. The allowance provided by this note, to enter the MODE of applicability with a single inoperable DG, takes into account the capacity and capability of the remaining AC sources and the fact that operation is ultimately limited by the Condition B Completion Time for the inoperable DG set.

(continued)

Information Only

BASES

ACTIONS
(continued)

B.4

Operation may continue in Condition B for a period that should not exceed 10 days.

In Condition B, the remaining OPERABLE DG set and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 10 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

Bases Insert 1

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite

(continued)

Information Only

BASES

ACTIONS

C.1 and C.2 (continued)

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

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

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

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

Bases Insert 20

D.1 and D.2

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

Operation may continue in Condition D for a period that should not exceed 24 hours.

Bases Insert 1

(continued)

Information Only

BASES

ACTIONS

D.1 and D.2 (continued)

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

E.1

With all or part of Train A DG set and Train B DG set inoperable, the capacity of the remaining standby AC sources is reduced depending on which combination of individual DGs is affected. Thus, with an assumed loss of offsite electrical power, standby AC sources may be insufficient to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

With all or part of each train of DG sets inoperable, operation may continue for a given unit for different periods of time depending on the combination of individual DGs that are inoperable. The length of time allowed increases with decreasing severity in the combinations of inoperable DGs. ~~One set must be restored to operable status in 2 hours if DGs 1-2A, 1C, and 1B on Unit 1 or DGs 1-2A, 1C, and 2B on Unit 2 are inoperable.~~ Operability of one set must be restored in 8 hours if DGs 1-2A and 1B on Unit 1 or DGs 1-2A and 2B on Unit 2 are inoperable. Operability of one set must be restored in 24 hours if DGs 1C and 1B on Unit 1 or DGs 1C and 2B on Unit 2 are inoperable.

Bases Insert 1

(continued)

Information Only

BASES

ACTIONS (continued)

F.1

Condition F provides the default Required Actions for the Conditions which address two inoperable offsite circuits or two inoperable DG sets. If the inoperable AC Sources cannot be restored to OPERABLE status within the applicable Completion Time, Required Action F.1 specifies that the unit be placed in MODE 3 within 6 hours. Once shut down, the unit is in a more stable condition and the time allowed to remain in MODE 3 is ultimately limited by the Required Actions and Completion Times applicable to a single inoperable AC Source based on the time that an AC Source initially became inoperable. In addition, the Required Actions applicable to one inoperable DG set or offsite circuit would remain applicable until both inoperable DG sets or offsite circuits are restored to OPERABLE status or the unit is placed in a MODE in which the LCO does not apply (MODE 5). The allowed Completion Times are reasonable to reach the required unit conditions from full power in an orderly manner and without challenging plant systems.

G.1

The sequencer(s) B1F, B2F, B1G, and B2G are an essential support system to both the offsite circuit and the DG associated with a given ESF bus. Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus. Therefore, loss of an ESF bus sequencer affects every major ESF system in the train. The 12 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.

Bases Insert 1

Bases Insert 22

I

~~H~~.1 and ~~H~~.2

If the inoperable AC electric power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is reduced. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 13). In MODE 4 the Steam Generators and Residual Heat Removal System are available

(continued)

ACTIONS

I

~~H~~.1 and ~~H~~.2 (continued)

to remove decay heat, which provides diversity and defense in depth. As stated in Reference 13, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action ~~H~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

~~I.1~~

~~Condition I corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. This condition exists when any combination of sources from the categories in LCO 3.8.1 totaling three or more are not OPERABLE. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.~~

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.108 (Ref. 9), as addressed in the FSAR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.19 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. This surveillance would also be applicable after any modifications which could affect DG interdependence.

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 1, 1971.
4. FSAR, Chapter 6.
5. FSAR, Chapter 15.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.108, Rev. 1, August 1977.
10. (Not used)
11. IEEE Standard 308-1971.
12. NEMA MG1-1967.
13. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.
14. (Add reference to SE here.)

Information Only

BASES

ACTIONS

A.1

Condition A represents one train of Auxiliary Building DC electrical power with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution system train.

~~[For Unit 1 only for cycle 19] The second Completion time for Condition A represents the 1B train of Auxiliary Building DC electrical power subsystem due to an inoperable battery. With the 1B Auxiliary Building battery inoperable, the DC bus is being supplied by the OPERABLE battery charger. Any event that results in a loss of the AC bus supporting the battery charger will also result in the loss of DC to that train. Recovery of the AC bus, especially if it is due to a loss of offsite power, will be hampered by the fact that many of the components necessary for the recovery (e.g., diesel generator control and field flash, AC load shed and diesel generator output breakers, etc.) rely upon the battery. The 12 hour limit allows sufficient time to effect restoration of the inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than 2.02 volts, etc.) are identified in Specifications 3.8.4, 3.8.5, and 3.8.6 together with additional specific completion times.~~

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power subsystems with attendant loss of ESF functions, in the case of the Auxiliary Building DC power subsystem, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 8) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the Auxiliary Building DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

Bases Insert 1

(continued)

Information Only

BASES

ACTIONS (continued)

B.1 and D.1

Conditions B and D represent one Auxiliary Building or SWIS DC electrical power subsystem with connection resistance not within the specified limit. Consistent with the guidance in IEEE-450, connection resistance not within the limit is an indication that the affected battery requires attention to restore the resistance to within the limit but is not a basis on which to declare the battery inoperable. Therefore, the 24 hour Completion Time allowed to restore the battery connection resistance to within the required limit is a reasonable time considering that variations in connection resistance do not mean the battery is incapable of performing its required safety function, but is an indication that the battery requires maintenance.

Bases Insert 1

C.1 and C.2

If the inoperable Auxiliary Building DC electrical power subsystem cannot be restored to OPERABLE status or the connection resistance restored to within the limit within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is reduced. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 11). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 11, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the

(continued)

Information Only

BASES

ACTIONS

C.1 and C.2 (continued)

Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If a required SWIS DC electrical power subsystem is inoperable or the connection resistance is not restored to within the limit and the associated Completion Time has expired, the Service Water System train supported by the affected SWIS DC electrical power subsystem must be declared inoperable. The capability of the affected SWIS DC electrical power subsystem to fully support the associated train of Service Water is not assured. Therefore, consistent with the definition of OPERABILITY, the associated train of Service Water must be declared inoperable immediately, thereby limiting operation in this condition to the Completion Time associated with the affected Service Water System train.

Bases Insert 23

Bases Insert 24

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is applying a voltage to the battery to maintain it in a fully charged condition during normal operation. The float voltage of 2.2 V per cell or 132 V overall is higher than the nominal design voltage of 125 V and is consistent with the manufacturer's recommendations for maintaining a full charge. Verifying that terminal voltage is ≥ 127.8 V provides assurance that the average of all cell voltages is maintained greater than 2.13 V. Maintaining float voltage at the higher value of 2.2 V per cell prolongs cell life expectancy. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODES 1, 2, 3, or 4. Risk insights or deterministic methods may be used for this assessment.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-1971.
4. FSAR, Section 8.3.
5. None.
6. FSAR, Chapter 6.
7. FSAR, Chapter 15.
8. Regulatory Guide 1.93, December 1974.
9. IEEE-450-1980.
10. Regulatory Guide 1.32, February 1972.
11. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.
12. (Add reference to SE here.)

Information Only

BASES

LCO (continued)

The intent of this Note is to limit the number of inverters that may be disconnected. Only those inverters associated with the single battery undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries, regardless of the number of inverters or unit design.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

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

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

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters — Shutdown."

ACTIONS

A.1

With a required inverter inoperable, its associated AC vital bus becomes inoperable until it is re-energized from its Class 1E CVT.

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems — Operating." This ensures that the vital bus is re-energized within 8 hours. The associated static transfer switch normally provides a bumpless transfer of power to the alternate AC source (Class 1E CVT).

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus

Bases Insert 1

(continued)

Information Only

BASES

ACTIONS

A.1 (continued)

is powered from its constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

Bases Insert 25

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

C

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is reduced. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 4, the steam turbine driven Auxiliary Feedwater Pump must be available to remain in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action ~~B~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 8.

2. FSAR, Chapter 6.

3. FSAR, Chapter 15.

4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 2010.

5. (Add reference to SE here.)

ACTIONS

A.1 and A.2 (continued)

remain energized via the Unit 1 or Unit 2 4160V H bus to which the 1C DG is aligned during a design basis accident. This will also ensure the 1C DG is unavailable to energize the affected unit. Therefore, consistent with the definition of OPERABILITY, the 1C DG must be declared inoperable for the affected unit.

B.1

If the Required Action and associated Completion Time of Condition A cannot be met, the power supply to the Unit 1 Service Water (SW) System automatic turbine building isolation valves (MOVs 515 and 517) will be unavailable following a design basis accident, so these valves must also be declared inoperable. Required Action A.2 will still apply, so the 1C DG must also be declared inoperable.

C.1 and C.2

With the shared Load Center 1-2R inoperable for reasons other than Condition A or Condition B, the Unit 1 Service Water (SW) System automatic turbine building isolation valves (MOVs 515 and 517) and the 1C DG must be declared inoperable immediately. The load center provides power to Unit 1 MOVs 515 and 517 and the 1C DG auxiliary systems. Therefore, consistent with the definition of OPERABILITY, these loads must be declared inoperable immediately.

D.1

With one or more required AC buses, load centers, motor control centers, or distribution panels, except AC vital buses, inoperable for reasons other than Condition A, B, or C, and a loss of safety function has not yet occurred, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety function necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

Bases Insert 1

(continued)

BASES

ACTIONS

D.1 (continued)

Condition D worst scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

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

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

E.1

With one or more AC vital buses inoperable, and a loss of safety function has not yet occurred, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital bus must be restored to OPERABLE status within 8 hours by powering the bus from the associated inverter via inverted DC or Class 1E constant voltage transformer.

Bases Insert 1

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

(continued)

BASES

ACTIONS

E.1 (continued)

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 8 hours if declared inoperable, is acceptable because of:

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

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

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

The 8 hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

F.1

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

Bases Insert 1

Condition F represents one train without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is,

BASES

ACTIONS

F.1 (continued)

therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

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

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

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

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

Bases Insert 26

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

H

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

If the inoperable distribution subsystem(s) addressed by Conditions D, E, ~~or~~ F, or G cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is reduced. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Remaining within the applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 the Steam Generators and Residual Heat Removal System are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference 4, the steam turbine driven Auxiliary Feedwater Pump must be available to remain

Information Only

H

BASES

ACTIONS

~~G~~.1 and ~~G~~.2 (continued)

in MODE 4. Should Steam Generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

H

Required Action ~~G.~~2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

I

~~H~~.1

With one SWIS DC electrical power distribution subsystem inoperable, the Service Water System train supported by the affected SWIS DC electrical power distribution subsystem must be declared inoperable. The capability of the affected SWIS DC electrical power distribution subsystem to fully support the associated train of Service Water is not assured. Therefore, consistent with the definition of OPERABILITY, the associated train of Service Water must be declared inoperable immediately, thereby limiting operation in this condition to the Completion Time associated with the affected Service Water System train.

~~I.1~~

~~With two trains with inoperable distribution subsystems that result in a loss of safety function, adequate core cooling, containment OPERABILITY and other vital functions for DBA mitigation would be compromised, and immediate plant shutdown in accordance with LCO 3.0.3 is required~~.

Information Only

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required AC, DC, and AC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Any change in the components being tested by this SR will require reevaluation of STI Evaluation Number 558904 in accordance with the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Chapter 6.

2. FSAR, Chapter 15.

3. Regulatory Guide 1.93, December 1974.

4. WCAP-16294-NP-A, Rev. 1, “Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs,” June 2010.

5. (Add reference to SE here)

Bases INSERT 1

Alternatively, a Completion Time can be determined using the Risk Informed Completion Time Program (Ref. xx).

Bases INSERT 2

This Condition is modified by two Notes. The first Note states it is not applicable when a Pressurizer Safety Valve is intentionally made inoperable. This Condition is not intended for voluntary removal of systems or components which would result in a loss of safety function. The Condition is only intended for a situation where a pressurizer safety valve is found inoperable. The second Note indicates the parts of Section 5.5.20, “Risk Informed Completion Time Program”, which are applicable to this LCO Condition. The Risk Informed Completion Time for this LCO Condition can be no longer than 24 hours.

Bases INSERT 3

For Required Action F.2, a Completion Time could also be determined using the Risk Informed Completion Time Program.

Bases INSERT 4

This Condition is modified by two Notes. The first Note states it is not applicable when a second block valve is intentionally made inoperable. This Condition is not intended for the voluntary removal of systems or components from service which would result in a loss of safety function. This Condition is intended only for the case of the second block valve being found inoperable. The second Note indicates the parts of Section 5.5.20, “Risk Informed Completion Time Program”, which are applicable to this LCO Condition. The Risk Informed Completion Time for this LCO Condition cannot exceed 24 hours.

Bases INSERT 5

C.1

With two or more accumulators inoperable for reasons other than boron concentration out of limits, the Required Action is to restore sufficient accumulators to OPERABLE status within 1 hour or, in accordance with the Risk Informed Completion Time Program, to regain the safety function. The Condition is modified by two Notes. The first Note states that the Condition is not applicable when two or more accumulators are intentionally made inoperable. The Required Action is not intended for voluntary removal of redundant components from service. The Required 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. The second Note indicates the parts of Section 5.5.20, “Risk Informed Completion Time Program”, which are applicable to this LCO Condition. The Risk Informed Completion Time for this LCO cannot exceed 24 hours.

Bases INSERT 6

Alternatively, a Completion Time may be determined using the Risk Informed Completion Time Program. However, a Risk Informed Completion Time may not be used for an inadequate water volume.

Bases INSERT 7

This Condition is modified by two Notes. The first Note states it is not applicable when the RWST is intentionally made inoperable. This Condition is not intended for voluntary removal of redundant systems or components from service. It is only intended for when the RWST is found inoperable. The second Note indicates the parts of Section 5.5.20, "Risk Informed Completion Time Program", which are applicable to this LCO Condition. The Risk Informed Completion Time for this LCO Condition can be no longer than 24 hours.

Bases INSERT 8

Condition B is modified by three Notes. The first Note states the Condition is only applicable to penetrations with two containment isolation valves. The second Note states the Condition is not applicable when the second containment isolation valve is intentionally made inoperable. The Condition is not intended for voluntary removal of removal of systems or components from service. The Condition is only intended for situations where the second containment isolation valve is found inoperable when the first containment isolation was inoperable for any reason, or when both isolation valves are simultaneously found inoperable. The third Note indicates those parts of Section 5.5.20 that are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 8a

The second Note states that the Condition is not applicable when the containment isolation valve is intentionally mode inoperable. The Condition is not intended for voluntary removal of systems or components from service. The Condition is only intended for situations where the containment isolation valve is found to be inoperable. The third Note indicates those parts of Section 5.5.20 that are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 9

B.1

With two containment spray trains inoperable, at least one containment spray train must be returned to OPERABLE status within 1 hour. Alternatively, a Completion Time can be determined using the Risk Informed Completion Time Program. Condition B is modified by two Notes. The first states that the Condition is not applicable when the second containment spray train in intentionally made inoperable. The Condition is not intended for voluntary removal of systems or components from service. The Condition is only intended for situations where the second containment spray train is found inoperable when the first spray system was already inoperable for any reason, or for when two containment spray systems are discovered inoperable at the same time. The second Note indicates the parts of Section 5.5.20 that are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 10

one containment spray or cooling unit must be restored to OPERABLE status within 1 hour. Alternatively, a Completion Time can be determined using the Risk Informed Completion Time Program. The Condition is modified by two Notes. The first Note states that the Condition is not applicable when the third containment cooling or spray train is intentionally removed from service. The Condition is not intended for the voluntary removal of systems or components from service. The Condition is only intended for situations where the third containment cooling or spray train is removed from service, and two other cooling or spray trains were Inoperable for any reason. The Condition may also be used when any combination of three containment cooling or spray trains are found inoperable at the same time. The second Note indicates the parts of Section 5.5.20 which are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 11

If one containment cooling or spray train cannot be returned to OPERABLE status within the required Completion Time, the unit must be placed in Mode 3 within 6 hours and in Mode 5 within 36 hours.

Bases INSERT 12

Condition B is modified by two Notes. The first one states that the Condition is not applicable when the second MSIV in a steam line is intentionally made inoperable. The Condition is not intended for the voluntary removal of systems or components from service. It is intended when the second MSIV is discovered inoperable when the first MSIV is inoperable for any reason. The second Note indicates those portions of Section 5.5.20 that are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 13

B.1

Required Action B.1 is applicable when there are two ARV lines inoperable. In this case, action must be taken to restore one ARV line to OPERABLE status. The 24 hour Completion time is reasonable because one ARV line is still available to conduct a cooldown following a SGTR event, the Steam Dump System and the MSSVs are available, and the low probability of an event occurring during this period that would require the ARV lines.

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

C.1

With all three ARV lines inoperable, a cooldown following a SGTR event cannot be conducted through the ARV lines. Consequently, at least one ARV line must be returned to OPERABLE status within 1 hour.

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

Condition C is modified by two Notes. The first Note states that it is not applicable when the third ARV line is intentionally removed from service. The Condition is not intended for voluntary removal of systems or components from service; it is intended only for situations where two ARV lines are inoperable for any reason, and the third line is intentionally made inoperable. The second Note describes which parts of Section 5.5.10 are applicable to this Condition.

The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 14

C.1

If two AFW trains are inoperable, the Required Action is to restore the inoperable AFW trains to OPERABLE status within 1 hour to regain a method of decay heat removal. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one train. Alternatively, a Risk Informed Completion Time can be determined.

The Condition is modified by two Notes. The first Note states it is not applicable when the second AFW train is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Condition is intended only when the second AFW train is found inoperable with one AFW train already inoperable for any reason, or if two AFW trains are discovered inoperable at the same time. The second Note indicates the parts of Section 5.5.20, “Risk Informed Completion Time Program”, which are applicable to this LCO Condition. The Risk Informed Completion Time for this LCO Condition can be no longer than 24 hours.

Bases INSERT 15

Condition A is modified by two Notes. The first Note states it is not applicable when the CST is intentionally removed from service. The Condition is not intended for voluntary removal of systems or components from service. The Condition is intended only when the CST is discovered inoperable. The second Note indicates the parts of Section 5.5.20, “Risk Informed Completion Time Program”, which are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 16

B.1

With both trains of CCW inoperable, the heat load capacity of the CCW system is seriously degraded such that the system may be incapable of providing an adequate heat sink for normal and accident conditions. Consequently, one hour is provided to restore the CCW trains to OPERABLE status. Alternatively, a Completion Time can be determined using the Risk Informed Completion Time Program.

The Condition is modified by two Notes. The first Note states it is not applicable when the second CCW train is intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Condition is intended only when the second CCW is found inoperable with one CCW train already inoperable, or if two CCW trains are discovered simultaneously inoperable. The second Note indicates the parts of Section 5.5.20, "Risk Informed Completion Time Program", that are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 17

B.1

With both SWS trains inoperable , the SWS may be incapable of providing an adequate heat sink for safety related components during design basis accidents and transients. Consequently, one hour is provided to restore the SWS train to OPERABLE status. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by two Notes. The first Note states it is not applicable when the second SWS train is intentionally made inoperable. This Condition is not intended for voluntary removal of redundant systems or components from service. The Condition is intended only when the second SWS train is discovered inoperable when the first train is already inoperable for reason. The Condition may also be used if both SWS trains are discovered inoperable simultaneously. The second Note indicates those portions of Section 5.5.20, "Risk Informed Completion Time Program", that are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 18

… one CRACs train must be returned to OPERABLE status within 1 hour. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Condition is modified by two Notes. The first Note states that the Condition is not applicable when the second CRACs train is intentionally removed from service. The Condition is not intended for the voluntary removal of redundant systems and components from service. Rather it is intended for when the second CRACs train is discovered inoperable when the first CRACs train is already inoperable for any reason. The second Note indicates the parts of Section 5.5.20, "Risk Informed Completion Time Program", that are applicable to this LCO Condition.

The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 19

F.1

If one CRACs train cannot be returned to OPERABLE status within the required Completion Time, the unit must be placed in Mode 3 within 6 hours and in Mode 5 within 36 hours.

Bases INSERT 20

Alternatively, for Condition C.2, a Completion Time may be determined in accordance with the Risk Informed Completion Time Program.

Bases INSERT 21

With two trains of the same ESF Room Cooler subsystems inoperable, the ability to cool the room housing ESF equipment sufficiently is jeopardized. The system may be rendered incapable of performing its accident mitigation function. Consequently, 1 hour is provided to restore one cooler to OPERABLE status. Alternatively, a Completion Time can be determined using the Risk Informed Completion Time Program. (Ref. 3).

The Condition is modified by two Notes. The first Note states that this Condition is not applicable when a second ESF train is intentionally made inoperable, and a first ESF train is already inoperable for any reason. The Condition is not intended for voluntary removal of redundant equipment from service. The Condition may also be used when two ESF Room Cooler subsystems from the same system are found inoperable simultaneously. The second Note indicates the parts of Section 5.5.20, "Risk Informed Completion Time Program', that are applicable to this LCO Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Bases INSERT 22

H.1

Condition H corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. This Condition exists when any combination of sources from the categories in LCO 3.8.1 totaling three or more are not OPERABLE. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, at least one AC source must be returned to Operable status within one hour or, alternatively, in accordance with the Risk Informed Completion Time Program.

The Condition is modified by two Notes. The first Note states that the Condition is not applicable when a third AC source is intentionally made inoperable, when two AC sources are already inoperable for any reason. The Condition is not intended for voluntary removal of redundant systems or components from service. The Condition may also be used when three or more AC sources are discovered inoperable simultaneously. The second Note indicates the parts of Section 5.5.20, "Risk Informed Completion Time

Program", that are applicable to this LCO Condition. The Risk Informed Completion for this Condition may not exceed 24 hours.

Bases INSERT 23

F.1

With two DC electrical power sources inoperable that result in a loss of power, the Required Action is to restore the required sources to OPERABLE status within one hour. The one hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of the required DC electrical power source(s). Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program (Ref. xx).

The Condition is modified by two Notes. The first Note states it is not applicable when the second DC source is intentionally made inoperable. The Condition is not intended for the voluntary removal of redundant systems or components from service. The Condition is only applicable if one DC electrical source is inoperable for any reason and a second DC source is found to be inoperable, or if two DC sources are found to be inoperable at the same time. The second Note indicates the parts of Section 5.5.20 that are applicable to this LCO Condition. The Risk Informed Completion Time for this Condition may be no longer than 24 hours.

Bases INSERT 24

G.1

If one DC source cannot be restored to OPERABLE status within the Completion Time of Condition F, the unit must be placed in Mode 3 within 6 hours and in Mode 4 within 12 hours.

Bases INSERT 25

B.1

With two or more inverters inoperable the Required Action is to restore the required inverters to OPERABLE status within one hour. The one hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of the required inverters. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program (Ref. xx).

The Condition is modified by two Notes. The first Note states it is not applicable when two or more required inverters are intentionally made inoperable. This Condition is not intended for voluntary removal of redundant systems and components from service. The Condition is only applicable if one required inverter is inoperable for any reason and a second inverter is discovered in operable, or if two inverters are simultaneously found inoperable. The second Note indicates the parts of Section 5.5.20. "Risk Informed Completion Time Program", which are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Information Only

Bases INSERT 26

G.1

With two trains with electrical distribution subsystems that result in a loss of safety function, the Required Action is to restore one train to OPERABLE status within one hour to restore safety function. The one hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of one train. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program (Ref. xx).

The Condition is modified by two Notes. The first Note states it is not applicable when two or more electrical distribution subsystems are intentionally made inoperable. This Condition is not intended for voluntary removal of redundant systems or components from service. The Condition is only applicable if one electrical power distribution subsystem is inoperable for any reason, and second subsystem is found to be inoperable, or if two electrical power distribution subsystems are simultaneously discovered inoperable. The second Note indicates those parts of Section 5.5.20, "Risk Informed Completion Time Program", which are applicable to this Condition. The Risk Informed Completion Time for this Condition may not exceed 24 hours.

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, “Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines”

Enclosure 1

List of Revised Required Actions to Corresponding PRA Functions

Table of Contents

1.0 Introduction and Summary

Section 4.0, Item 2 of the Final Safety Evaluation for NEI 06-09 (Revision 0-A, Reference 1) identifies the following License Amendment Request (LAR) content needed on applicable Technical Specifications (TSs), 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 Operations (LCO) and action requirements to which the Risk Informed Completion Time (RICT) Program 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, flowrate, etc., are consistent licensing basis assumptions (i.e., 10 CFR 50.46 ECCS flow rates) for each of the TS requirements, or an appropriate disposition or programmatic restriction will be provided.

This enclosure provides confirmation that the Farley Nuclear Plant (FNP) PRA models include the necessary scope of structures, systems, and components (SSCs) and their functions to address each proposed application of the RICT Program to the TS LCO Conditions. The scope of the comparison includes each of the TS LCO conditions and associated required actions applicable to RICT Program implementation at FNP Units 1 and 2.

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

- <u>Column "TS LCO Condition"</u>: Lists all of the LCOs and Condition statements within the scope of the RICT Program.
- <u>Column "SSCs Covered by TS LCO Condition"</u>: The SSCs addressed by each Action requirement.
- <u>Column "SSCs Modeled in PRA"</u>: Indicates whether the SSCs addressed by the TS LCO Condition are included in the PRA.

- Column "Function Covered by TS LCO Condition": A summary of the required functions from the design basis analyses.

- Column "Design Success Criteria": The function success criteria as documented in the Technical Specifications bases and/or FSAR.

- Column "PRA Success Criteria": The function success criteria modeled in the PRA, as specified in the referenced PRA documentation and verified in the PRA model files.

- 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 assessment of the core damage risk of the TS Condition as required by NEI 06-09 and PRA standards for Capability Category (CC) II.

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 difference between the PRA scope or PRA success criteria are described in the table. Scope differences are justified by identifying appropriate surrogate events which permit a risk assessment to be completed using the Configuration Risk Management Program (CRMP) tool for the RICT Program. Differences in success criteria typically arise due to the requirement in the PRA standards (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 CC II of the PRA standards as required by NEI 06-09 (Reference 1).

The calculated RICTs, provided in Table E1.2, demonstrate the effect on CDF and LERF for each individual condition to which the RICT Program applies (assuming no other SSCs modeled in the PRA outside the scope of the applicable TS LCO Condition are unavailable). These calculations were performed based on the use of separate zero-maintenance annual average PRA models which include the internal events PRA model, internal fire PRA model that reflects NFP-805 plant modifications, seismic bounding delta CDF/LERF values and main control room abandonment bounding delta CDF/LERF values. Use of the main control room abandonment bounding values may be discontinued in the future if the fire PRA models are revised to include detailed modeling of main control room abandonment risk contribution. In addition, the RICT calculations in Table E1.2 assume that a single SSC impacts the applicable TS LCO Condition for most cases; however, in some cases, more than one SSC was considered to impact the TS LCO Condition to ensure a more limiting case RICT can be generated for conditions that allow more than one train inoperable but do not meet the criteria for a loss of function. In such cases there are two entries for that LCO. These estimates are based on a Unit 1 model calculation and are considered applicable to Unit 2 for the purpose of providing an estimate due to the close similarity between the Unit 1 and Unit 2 models. The actual RICT values during program implementation will be calculated based on the actual unit and plant configuration and the on-record version of the CRMP model available which represents the as-built and as-operated plant, as required by NEI 06-09 and the NRC Safety Evaluation. For the values presented in the “RICT Calculated” column of Table E1.2, the equipment removed from service for the calculation is the piece of equipment associated with the applicable LCO Condition.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.4.10 Pressurizer Safety Valves (PSV) A One pressurizer safety valve inoperable. TS Loss of Function (TS LOF) ------------NOTES----------- 1. Not applicable when a PSV is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c2, c3, d, e, f, g and h	3 PSVs	Yes	Prevent RCS pressure from exceeding safety limit	3 of 3 PSVs	Same as Design Success Criteria for limiting transient (ATWT with initial reactor power > 40%)	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria. TS LOF PRA Functionality Requirements: • 1 PSV Inoperable requires 3 PSVs PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in P RA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.4.11 Pressurizer Power Operated Relief Valves (PORV) B One PORV inoperable and not capable of being cycled.	2 PORVs	Yes	Depressurize the RCS in certain transients	1) 1 of 2 PORVs for opening. 2) 2 of 2 PORVs must not have excessive leakage.	1) Same or more restrictive 2) Function not specifically modeled	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria and in some cases are more restrictive when the PORVs are used to mitigate some beyond design basis scenarios The Function 2 success criteria of “2 of 2 PORVs must not have excessive leakage” have no consequence on the likelihood of mitigating a worst case ATWT event. As a result, the success criteria in the PRA are consistent with the design basis criteria.
3.4.11 Pressurizer Power Operated Relief Valves (PORV) C One Block Valve inoperable	2 Block Valves	Yes	Isolate the flow path through a PORV with excessive leakage.	Associated block valve closed to prevent leakage	Same as Design Success criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.4.11 Pressurizer Power Operated Relief Valves (PORV) F Two blocks valves inoperable. ------------NOTES----------- 1. Not applicable when the second block valve is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h	See LCO Condition 3.4.11.C					See LCO Condition 3.4.11.C. TS LOF PRA Functionality Requirements: : • 2 Block Valves Inoperable requires at least 1 Block Valve PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.5.1 Accumulators C Two or more Accumulators inoperable for reasons other than boron concentration not within limits (TS LOF) ------------NOTES----------- 1. Not applicable when two or more ECCS accumulators are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g and h.	ECCS Accumulators	ECCS Accumu-lator valves as surrogate	Supply borated water to the reactor vessel during LOCA blowdown phase.	2 of 3 accumulators	For LLOCA and MLOCA accidents 2 of 2 Accumulators to 2 of 2 intact cold legs For SLOCA and Consequential LOCA 2 out of 3 Accumulators to 2 out of 3 cold legs.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria. TS LOF PRA Functionality Requirements: • 2 Accumulators Inoperable requires at least 1 Accumulator PRA Functional • 3 Accumulators Inoperable requires at least 2 Accumulators PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	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 ECCS – Operating A One or more trains inoperable.	3 Centrifugal charging pumps (CCPs) 2 RHR pumps 2 RHR heat exchangers	Yes	Provide core cooling and negative reactivity for: 1) LOCA 2) Rod Ejection Accident 3) Loss of secondary coolant accident 4) Steam Generator Tube Rupture	1 of 3 CCPs 1 of 2 RHR pumps	LHI (Low-Head Injection) 1 of 2 LHSI pumps deliver flow to 2 intact RCS CLs HHI (High-Head Injection) 1 of 3 CCPs deliver flow to 2 intact RCS CLs HLR (Hot Leg Recirculation) 1 of 2 LHSI pumps deliver flow to 1 intact RCS Hot Leg (HL) LHR (Low-Head Recirculation) 1 of 2 LHSI pumps deliver flow to 2 intact RCS CLs LTC (Long Term Cooling - HHR) 1 of 3 CCPs delivers flow to 2 intact RCS CLs LTC (Long Term Cooling – LHR) 1 of 2 RHR trains deliver flow to 2 intact RCS CLs SIT (SI Termination) Operator terminates CCPs and establishes normal charging	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria and in some cases mitigate some beyond design basis scenarios like SIT (SI Termination) where Operator terminates CCPs and establishes normal charging.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.5.4 Refueling Water Storage Tank B RWST inoperable for reasons other than Condition A. (TS LOF) ------------NOTES----------- 1. Not applicable when the RWST is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	RWST	Yes	Supply borated water to ECCS and Containment Spray during LOCA phase for 1) negative reactivity for reactor shutdown, and 2) core and containment cooling and containment depressurization	Reasons other than boron concentration limits and temperature limits met.	Same as Design Success Criteria	TS LOF PRA Functionality Requirements: • RWST is required to be PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.6.2 Containment Air Locks C One or more containment airlock doors open for reasons other than Conditions A or B.	Containment Airlock Doors	Yes	Control of Post-Accident Containment Leakage Rates	Post-Accident Containment Leakage Rates within limits	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria
3.6.3 Containment Isolation Valves A One or more penetration flow paths with one containment isolation valve inoperable except for purge valve penetration leakage not within limit. -----------NOTE------------ Only applicable to penetration flow paths with two containment isolation valves.	Two isolation devices	Yes	Isolate Containment within assumed time limits to prevent excessive RCS loss and establish containment pressure boundary post-accident	One Containment isolation valve closed within stroke time limits, if applicable.	Same as Design Success Criteria	The PRA does not explicitly model the impact of excessive stroke time. This condition can be addressed for the RICT Program by assuming the inoperable containment isolation valve(s) to be unavailable (failed open) in the PRA model if it is open. Therefore, this LCO condition can be evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.6.3 Containment Isolation Valves B One or more penetration flow paths with two containment isolation valves inoperable except for purge valve penetration leakage not within limit (TS LOF) ------------NOTES----------- 1. Only applicable to penetration flow paths with two containment isolation valves. 2. Not applicable when the second Containment Isolation valve is intentionally made inoperable. 3. The following Section 5.5.20 Constraints apply: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Condition 3.6.3.A TS LOF PRA Functionality Requirements: : • 2 Penetration Flow Paths Inoperable requires at least 1 Penetration Flow Paths PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: Yes • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None					

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.6.3 Containment Isolation Valves C One or more penetration flow paths with one containment isolation valve inoperable. (TS LOF) --------NOTES----- 1. Only applicable to penetration flow paths with one containment isolation valve and a closed system. 2. Not applicable when the second Containment Isolation valve is intentionally made inoperable. 3. The following Section 5.5.20 Constraints apply: parts b, c.2, c.3, d, e, f, g, and h.	Penetration flow paths with one isolation valve and a closed system	Yes	See LCO Condition 3.6.3.A	One Containment isolation valve closed within stroke time limits, if applicable. Closed system intact.	See LCO Condition 3.6.3.A	See LCO Condition 3.6.3.A TS LOF PRA Functionality Requirements: : • 2 Penetration Flow Paths Inoperable requires at least 1 Penetration Flow Paths PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: Yes • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.6.6 Containment Spray and Cooling Systems A One containment spray train inoperable.	2 Containment Spray Systems	Yes	Provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature and to reduce fission products	1 of 2 Containment Spray trains	Same as Design Success Criteria	The PRA models the containment heat removal function consistently with the DBA. However, the PRA does not model the fission product removal functions. Use of RICT for this TS Condition is contingent on the sufficiency and availability of the fission product removal functions.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.6.6 Containment Spray and Cooling Systems B Two containment spray trains inoperable (TS LOF) ------------NOTES----------- 1. Not applicable when the second containment spray train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g and h.	See LCO Condition 3.6.6.A					See LCO Condition 3.6.6.A TS LOF PRA Functionality Requirements: • One containment spray system is required to be PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.6.6 Containment Spray and Cooling Systems D One containment cooling train inoperable.	2 Containment cooling trains	Yes	Limits the ambient containment air temperature during normal unit operation to less than the design limit.	1 of two containment cooling trains	2 of 4 CCS Fan Coolers (FCs) The CCS functions during normal operations are not modeled but PRA modeling is more restrictive and supports DBA.	SSCs modeled in the PRA using a more restrictive success criteria for the DBA than the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria
3.6.6 Containment Spray and Cooling Systems E Two containment cooling trains inoperable.	See LCO Condition 3.6.6.D					See LCO Condition 3.6.6.D

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.6.6 Containment Spray and Cooling Systems G Any combination of three or more trains inoperable. (TS LOF) ------------NOTES----------- 1. Not applicable when three or more combinations of trains are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Condition 3.6.6.A and 3.6.6.D					See Condition LCO Condition 3.6.6.A and 3.6.6.D TS LOF PRA Functionality Requirements: • 2 Containment Spray trains and 1 containment cooling train Inoperable requires at least 1 containment spray train PRA Functional • 2 Containment cooling trains and 1 containment spray train Inoperable requires at least 1 containment cooling train PRA Functional • 2 Containment Spray trains and 2 containment cooling train Inoperable requires at least 1 containment spray train and 1 containment cooling train PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.7.2 Main Steam Isolation valves A One or more steam lines with one MSIV inoperable in MODE 1.	2 MSIVs per steam line	Yes	Isolate steam flow from the secondary side of the steam generators in a High Energy Line Break.	One MSIV closes in each steam line	(1) SGI (SG Isolation) for SSB: 1 of 2 MSIVs closed on all three SGs to prevent blowdown of the intact SGs (2) SGI (Ruptured SG Isolation) for SGTR: • 1 of 2 MSIVs closed on ruptured SG OR • 1 of 2 MSIVs on each of 2 intact SGs closed to prevent blowdown of the ruptured SG.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool and EOOS model. The success criteria in the PRA are consistent with the design basis criteria

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.2 Main Steam Isolation valves B One or more main steam lines with two MSIVs inoperable in MODE 1. (TS LOF) ------------NOTES----------- 1. Not applicable when second MSIV in a line is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Condition 3.7.2.A					See LCO Condition 3.7.2.A TS LOF PRA Functionality Requirements: • 1 main steam line with two MSIVs inoperable requires at least 1 MSIV PRA Functional in each steamline • 2 main steam lines with two MSIVs inoperable requires at least 1 MSIV PRA Functional in each steamline • 3 main steam lines with two MSIVs inoperable requires at least 1 MSIV PRA Functional in each steamline • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.7.4 Atmospheric Relief Valves A One required ARV line inoperable.	3 Atmospheric Relief Valves	Yes	Cools the unit to RHR entry conditions if the preferred heat sink via the steam dump system to the main condenser becomes unavailable.	One ARV remains available to conduct a unit cooldown following a SGTR.	Same as Design Basis Criteria except for ATWT conditions, then 4 of 4 ARV Lines.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria with exception noted below PRA SC differs from the DB SC, the PRA SC are judged to be more realistic and restrictive than those assumed in the DB analysis.
3.7.4 Atmospheric Relief Valves B Two required ARV lines inoperable.	See LCO Condition 3.7.4.A					

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.4 Atmospheric Relief Valves C Three required ARV lines inoperable (TS LOF) ------------NOTES----------- 1. Not applicable when the third ARV in a line is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Condition 3.7.4.A.					See LCO Condition 3.7.4.A. TS LOF PRA Functionality Requirements • 3 ARVs lines inoperable requires 1 ARV lines to be PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: OPERATOR FAILS TO LOCALLY OPEN ATMOS RELIEF VLVS ON LOSS OF SUPPORT • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.7.5 Auxiliary Feedwater System A Turbine driven AFW train inoperable due to one inoperable steam supply	2 steam supplies	Yes	Provide a steam supply to the turbine driven auxiliary feedwater pump.	1 of 2 steam supplies available	Same as Design Basis Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria
3.7.5 Auxiliary Feedwater System B One AFW train inoperable for reasons other than Condition A.	2 motor driven auxiliary feedwater pumps, and 1 turbine driven.	Yes	Supply feedwater to the steam generators to remove heat.	2 of 3 AFW pumps	1 of 3 except for ATWT conditions, where more restrictive criteria of 3 of 3 are applied	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are based on a realistic analysis and for all initiators except ATWT are less restrictive than the design basis criteria, and more conservative for mitigation of beyond design basis ATWT scenarios.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.6 Condensate Storage Tank A CST Inoperable (LOF) ------------NOTES----------- 1. Not applicable when the CST is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, and g and h.	One condensate storage tank	Yes	Provides a safety grade source of water to the Steam Generators. Also provides a passive flow of water to the Auxiliary Feedwater (AFW) System.	CST Operable	CST available OR Plant Service Water suction source to AFW pumps available.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. Since this is a TS LOF Condition, PRA parameter success criteria are overridden by design basis parameters for the purpose of establishing PRA functionality. An NRC approval is sought as part of this LAR submittal to credit use of plant service water as modeled in the PRA as an alternate source of water to recover degraded CST design basis parameters for establishing PRA Functionality. The PRA success criteria are consistent with the design basis criteria. LOF PRA Functionality Requirements: • 1 CST Inoperable requires 1 CST OR Plant Service Water suction source to AFW pumps available to recover degraded CST design basis parameters. • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: Failure of OPERATOR to align SW TO AFW Pump Suction • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.7.7 Component Cooling Water A One CCW train inoperable	2 trains of CCW each with one full capacity pump.	Yes	The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient.	One of two CCW trains	Same as Design Success Criteria and the initial containment temperature assumed in the PRA Success Criteria analysis is 125°F, max design basis containment sump temp assumed is 132.8°F.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria for the number of pump trains required. The inlet sump temperature in the PRA is a function of the realistic accident progression conditions experienced for the accident sequence being analyzed by MAAP. Realistic success criteria are used consistent with the PRA standards for CC II.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.7 Component Cooling Water B Two CCW trains inoperable. (TS LOF) ------------NOTES----------- 1. Not applicable when second CCW train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Condition 3.7.7.A.					See LCO Condition 3.7.7.A TS LOF PRA Functionality Requirements: : • 2 CCW trains Inoperable requires at least 1 CCW train PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function: • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.7.8 Service Water System A One SWS Train inoperable	2 SWS trains each consisting of 2 50% capacity pumps and 1 50% capacity shared pump.	Yes	Provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient.	One SWS train, in conjunction with the CCW System and a 100% capacity containment cooling system.	(1) 1 SW train with 1 SW pump per train for non-LOSP/non-SI conditions, (2) 1 SW train with 1 SW pump per train for LOSP prior to the need for RHR cooling and if the dilution bypass valves are not open, and (3) 1 SW train with 2 SW pumps per train for SI conditions.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are less restrictive than the design basis criteria for non-LOSP/non-SI conditions, but are consistent with the design basis criteria for other conditions, and are more realistic and consistent with the PRA standards for CC II which requires use of realistic analysis to support a RI application.
3.7.8 Service Water System B Two SWS Trains inoperable. (TS LOF) ------------NOTES----------- 1. Not applicable when the second SWS train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Condition 3.7.8.A					See LCO Condition 3.7.8.A TS LOF PRA Functionality Requirements: : • 2 Service Water trains Inoperable requires at least 1 Service Water train PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function: • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.11 Control Room Air Conditioning System (CRACS) E Two CRACs trains inoperable in MODE 1, 2, 3, or 4. (TS LOF) ------------NOTES----------- 1. Not applicable when second CRACS train is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	Two independent and redundant trains of the Control Room Air Conditioning System	Yes	Provides temperature control for the FNP common control room by maintaining an adequate control room temperature for 30 days of continuous occupancy.	One CRACS train	Not Modeled- Documented in PRA basis as not needed to prevent to core damage.	See LCO Condition 3.7.11.A TS LOF PRA Functionality Requirements: • 2 CRACS trains requires at least 1 CRACS train to be PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: examples of simple and uncomplicated actions include opening doors and starting the opposite train cooler with at least 16 hours available to prevent a reactor trip on loss of control room cooling. • SSCs not modeled in PRA for Function: None
3.7.19 Engineered Safety Features (ESF) Room Coolers A One required ESF Room Cooler subsystem Train inoperable.	Two ESF Room Cooler and Safety-Related Chiller Trains		Room cooling for ESF equipment provided by ESF Room Coolers. The Room Coolers are divided into subsystems and each subsystem has two 100% capacity trains.	1 of 2 trains.	Same as Design Success Criteria	Charging Pump A and C belong to Train A and B, respectively. Charging Pump B is the swing pump and can align to either train.. The swing pump and its associated cooler can be powered from either Train A or B. ESF Room Cooler Subsystems are: • Motor Driven Auxiliary Feedwater (MDAFW) Pump Rooms • Charging Pump Rooms • Containment Spray (CS) Pump Rooms • Component Cooling Water (CCW) Pumps Room • Auxiliary Building DC Switchgear / Battery Charger Rooms • Load Control Center (LCC) Rooms (LCC D and E Rooms) The ESF room coolers are considered support equipment for ESF equipment in the above rooms with the exception of the CCW Pumps Room CCW and Load Centers room cooling is not required to prevent core damage per PRA, as a result a 30 day back applies.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.19 Engineered Safety Features (ESF) Room Coolers B Two trains of the same ESF Room Cooler subsystem inoperable (TS LOF) ------------NOTES----------- 1. Not applicable when the second ESF Room Cooler is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g and h.	See LCO Condition 3.7.19.A					See LCO Condition 3.7.19.A TS LOF PRA Functionality Requirements: • 2 ESF room cooler trains of the same subsystem requires at least 1 ESF room cooler train to be PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: Recovery actions for opening doors for rooms housing MDAFW pumps, Charging pumps, CS pumps and DC Switchgears • Manual actions not credited in PRA for Function: For CCW and Load centers rooms no operator actions are assumed in the PRA • SSCs not modeled in PRA for Function: None
3.8.1 AC Sources – Operating A One required offsite circuit inoperable	Breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es).	Yes	Transmit power from offsite transmission network to onsite Class 1E ESF buses	1 of 2 circuits.	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.8.1 AC Sources – Operating B One DG set inoperable.	2 DG Sets, each set comprised of 2 DGs.	Yes	Upon loss of preferred power, supply ESF loads in time to mitigate consequences of a DBA	1 of 2 DG Sets.	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	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 AC Sources – Operating C Two required offsite circuits inoperable.	See LCO Condition 3.8.1.A					See LCO Condition 3.8.1.A
3.8.1 AC Sources – Operating D One required offsite circuit inoperable. <u>AND</u> One DG set inoperable.	Breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es), and 2 sets of DGs, each set comprised of 2 DGs.	See LCO Conditions 3.8.1.A and 3.8.1.B				See LCO Conditions 3.8.1.A and 3.8.1.B
3.8.1 AC Sources – Operating E Two DG sets inoperable	See LCO Condition 3.8.1.A					See LCO Conditions 3.8.1.A and 3.8.1.C
3.8.1 AC Sources – Operating G One Automatic Load Sequencer inoperable	2 sequencers	Yes	1) Provides a pre-determined sequence of loading the DGs, and 2) Also actuates the ESF loads on the offsite circuits when offsite power is available.	1 of 2 sequencers for both functions 1 and 2	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	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 AC Sources H Three or more required AC Sources inoperable. (TS LOF) ------------NOTES----------- 1. Not applicable when three or more AC sources are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Conditions 3.8.1.A and 3.8.1.B					See LCO Conditions 3.8.1.A and 3.8.1.B TS LOF PRA Functionality Requirements: • 3 AC Sources Inoperable: 2 DG trains and 1 offsite AC source inoperable (1 offsite source operable) requires at least 1 offsite AC source or 1 DG train PRA Functional • 3 AC Sources Inoperable: 1 DG train (1 offsite AC operable) and 2 offsite AC sources inoperable requires at least 1 DG trains or 1 offsite AC source PRA Functional • 4 AC sources inoperable: 2 DG Trains and 2 Offsite AC sources Inoperable requires at least 1 DG train and 1 offsite source PRA Functional; OR • 4 AC sources inoperable: 2 DG Trains and 2 Offsite AC sources Inoperable requires at least 2 DG trains PRA Functional; OR • 4 AC sources inoperable: 2 DG Trains and 2 Offsite AC sources Inoperable requires at least 2 Offsite AC sources PRA Functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
3.8.4 DC Sources – Operating A One Auxiliary Building DC electrical power subsystem inoperable	2 trains of Auxiliary DC system	Yes	Supplies DC power to various ESF systems throughout the plant.	1 of 2 trains	Same as Design Success Criteria with the exception that PRA models reactor trip on loss of AB DC train.	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria except that PRA models reactor trip on loss of AB DC train. This is consistent with the plant practice of initiating a reactor trip on loss of AB DC train
3.8.4 DC Sources – Operating B One Auxiliary Building DC electrical power subsystem with battery connection resistance not within limit.	See LCO Condition 3.8.4.A					

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.8.4 DC Sources – Operating D One required SWIS DC electrical power subsystem battery connection resistance not within limit.	Four 125 VDC batteries with battery chargers (Shared between the two units).	Yes	Provide a reliable source of power for controls, power loads, annunciation and alarms	1 of 2 subsystems.	1 of 2 trains supporting 2 of 2 SW Pumps per train	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria except that PRA additionally requires 2 of 2 SW pumps per train.
3.8.4 DC Sources – Operating F Two or more DC electrical subsystems inoperable that result in a loss of function (TS LOF) ------------NOTES----------- 1. Not applicable when second DC power electrical subsystem is intentionally removed from service. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Condition 3.8.4.A					See LCO Condition 3.8.4.A TS LOF PRA Functionality Requirements: • Two DC electrical subsystems inoperable requires at least one DC electrical power subsystem to be PRA functional • Three DC electrical subsystems inoperable requires at least 1 DC electrical power subsystem to be PRA functional • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None •
3.8.7 Inverters – Operating A One required inverter inoperable	4 Class 1E inverters	Yes	Provides reliable AC electrical power to the vital buses	One train with 2 of 2 inverters, (each train redundant).	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.8.7 Inverters - Operating B Two or more required inverters inoperable. (TS LOF) ------------NOTES----------- 1. Not applicable when second required inverter is intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Condition 3.8.7.A					See LCO Condition 3.8.7.A TS LOF PRA Functionality Requirements: • One required inverter in each train inoperable requires one train to have two inverters to be PRA functional • Two required inverters inoperable (both in one train) requires two inverters on the opposite train • Three required inverters inoperable (two on one train and one on opposite train) requires either one in the opposite train to be PRA functional. • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None •
3.8.9 Distribution Systems - Operating D. One or more AC electrical distribution subsystems inoperable for reasons other than Condition A., B, or C	Two trains each of AC Safety buses	Yes	Provide necessary power to ESF systems	1 of 2 AC trains	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.8.9 Distribution Systems - Operating E One or more AC Vital buses inoperable	Two AC Vital distribution panels per train	Yes	Provide necessary power to Essential Instrumentation.	1 Train with 2 of 2 distribution panels (each train redundant)	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.
3.8.9 Distribution Systems- Operating F One Auxiliary Building DC electrical power distribution subsystem inoperable	Two DC Distribution Panels per train	Yes	Provides a source of DC power for control and instrumentation during normal conditions and design basis events.	2 of 2 Distribution Panels in one train. (each train redundant)	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. The success criteria in the PRA are consistent with the design basis criteria.

Table E1.1
Revised TS LCO Conditions to Corresponding PRA Functions

TS LCO Condition (Note 1)	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.8.9 Distribution Systems -Operating G Two trains with inoperable distribution subsystems that result in a loss of function. (TS LOF) ------------NOTES----------- 1. Not applicable when two or more electrical power distribution trains are intentionally made inoperable. 2. The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g, and h.	See LCO Conditions 3.8.9.A thru 3.8.9 C					See LCO Conditions 3.8.9.A thru 3.8.9 C LOF PRA Functionality Requirements: : • 2 **trains inoperable requires at least one 1 train PRA Functional** • Design basis criteria for parameters overrides PRA Success criteria for parameters for Function • Manual actions credited in PRA for Function: None • Manual actions not credited in PRA for Function: None • SSCs not modeled in PRA for Function: None
Note 1: The Technical Specification Condition as described in this table may not exactly match one-to-one with the Condition as described in the FNP Technical Specifications. In some cases, the Condition description is revised to add information to enhance the clarity pertinent to the context of this table. Refer to Attachment 1 for the exact description of Technical Specifications Condition.						

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.4.10 Pressurizer Safety Valves A. One pressurizer safety valve inoperable. TS Loss of Function (TS LOF)	1PZRV8010A---D (PRA Functional) (PRESSURIZER SV 8010A FAILS TO OPEN DUE TO RANDOM FAULTS)	24 hrs. (24 hrs.))
3.4.11 Pressurizer Power Operated Relief Valves (PORV) B. One PORV inoperable and not capable of being cycled.	1PZAV444B----D (HARDWARE FAULTS OF PORV 444B)	30.0 (30.0)
3.4.11 Pressurizer Power Operated Relief Valves (PORV) C. One Block Valve inoperable	1PZMV8000B---K (PRESSURIZER PORV BLOCK VALVE 8000B FAILS TO CLOSE)	30.0 (30.0)
3.4.11 Pressurizer Power Operated Relief Valves (PORV) F. Two block valves inoperable	1PZMV8000A---K (PRESSURIZER PORV BLOCK VALVE 8000A FAILS TO CLOSE) and 1PZMV8000B---K (PRESSURIZER PORV BLOCK VALVE 8000B FAILS TO CLOSE)	30.0 (30.0)
3.5.1 Accumulators C. Two or more Accumulators inoperable for reasons other than boron concentrations not within limits (TS LOF)	1HHMV8808A---V (ACCUMULATOR 1A ISOLATION VALVE)	24 hrs. (24 hrs.)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.5.2 ECCS – Operating A. One or more trains inoperable.	1LHPMP001A---A (RHR/LHI PUMP P-001A FAILS TO START DUE TO RANDOM FAILURE)	14.3 (10.7)
3.5.2 ECCS – Operating A. One or more trains inoperable.	1HHPMP002A---A (CHG PUMP P002A FAILS TO START)	30.0 (25.0)
3.5.4 Refueling Water Storage Tank B. RWST inoperable for reasons other than Condition A. (TS LOF)	1SITKF16T501-R (PRA Functional) (RWST RUPTURES)	24 hrs. (24 hrs.)
3.6.2 Containment Air Locks C. One or more containment airlock doors open for reasons other than Conditions A or B	ADMN-PEN-NI (surrogate) (ADMINISTRATIVELY CONTROLLED PENETRATIONS NOT ISOLATED)	30.0 (5.3)
3.6.3 Containment Isolation Valves A. One or more penetration flow paths with one containment isolation valve inoperable except for purge valve penetration leakage not within limit	1CICVB13V038-K (CHECK VALVE QnB31V038 FAILS TO CLOSE)	30.0 (7.1)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.6.3 Containment Isolation Valves A. One or more penetration flow paths with one containment isolation valve inoperable except for purge valve penetration leakage not within limit	1CICVG21V204-K (CHECK VALVE QnG21V204 FAILS TO CLOSE)	30.0 (26.2)
3.6.3 Containment Isolation Valves B One or more penetration flow paths with two containment isolation valves inoperable except for purge valve penetration leakage not within limit	1CIAVB13V040-K (AOV QnB31V040 FAILS TO CLOSE (HARDWARE)) and 1CICVB13V038-K (CHECK VALVE QnB31V038 FAILS TO CLOSE)	30.0 (5.3)
3.6.3 Containment Isolation Valves C. One or more penetration flow paths with one containment isolation valve inoperable.	ADMN-PEN-NI (surrogate) (ADMINISTRATIVELY CONTROLLED PENETRATIONS NOT ISOLATED)	30.0 (5.3)
3.6.6 Containment Spray and Cooling Systems A. One containment spray train inoperable	Not Modeled since not needed for core damage prevention. 30 day back stop applies	30.0 (30.0)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.6.6 Containment Spray and Cooling Systems B. Two containment spray trains inoperable (TS LOF)	Not Modeled since not needed for core damage prevention. 30 day back stop applies	24 hrs. (24 hrs.)
3.6.6 Containment Spray and Cooling Systems D. One containment cooling train inoperable.	1FCMOH001D---F (FAN MOTOR D FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS) ,1FCMOH001C---F (FAN MOTOR C FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS)	30.0 (30.0)
3.6.6 Containment Spray and Cooling Systems E. Two containment cooling trains inoperable.	1FCMOH001D---F(FAN MOTOR D FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS),1FCMOH001C---F (FAN MOTOR C FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS),1FCMOH001A---F (FAN MOTOR A FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS),1FCMOH001B---F (FAN MOTOR B FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS)	15.5 (30.0)
3.6.6 Containment Spray and Cooling Systems G. Any combination of three or more trains inoperable (TS LOF)	1FCMOH001D---F(FAN MOTOR D FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS),1FCMOH001C---F (FAN MOTOR C FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS),1FCMOH001A---F (FAN MOTOR A FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS),1FCMOH001B---F (FAN MOTOR B FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS)	24 hrs. (24 hrs.)
3.6.6 Containment Spray and Cooling Systems G. Any combination of three or more trains inoperable (TS LOF)	1FCMOH001D---F (FAN MOTOR D FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS) ,1FCMOH001C---F (FAN MOTOR C FAILS TO SWITCH SPEEDS DUE TO RANDOM FAULTS)	24 hrs. (24 hrs.)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.7.2 Main Steam Isolation valves A. One or more steam lines with one MSIV inoperable in MODE 1.	1MSHV3369A---K (MSIV HV-3369A FAILS TO CLOSE DUE TO HARDWARE FAULTS), 1MSHV3369B---K, HV-3369B FAILS TO CLOSE DUE TO HARDWARE FAULTS), 1MSHV3369C---K (HV-3369C FAILS TO CLOSE DUE TO HARDWARE FAULTS),	30.0 (30.0)
3.7.2 Main Steam Isolation valves B. One or more main steam lines with two MSIVs inoperable in MODE 1. (TS LOF)	1MSHV3369A---K (MSIV HV-3369A FAILS TO CLOSE DUE TO HARDWARE FAULTS)	24 hrs. (24 hrs.)
3.7.2 Main Steam Isolation valves B. One or more main steam lines with two MSIVs inoperable in MODE 1. (TS LOF)	1MSHV3369A---K (MSIV HV-3369A FAILS TO CLOSE DUE TO HARDWARE FAULTS), 1MSHV3369B---K, HV-3369B FAILS TO CLOSE DUE TO HARDWARE FAULTS),	24 hrs. (24 hrs.)
3.7.2 Main Steam Isolation valves B. One or more main steam lines with two MSIVs inoperable in MODE 1. (TS LOF)	1MSHV3369A---K (MSIV HV-3369A FAILS TO CLOSE DUE TO HARDWARE FAULTS), 1MSHV3369B---K, HV-3369B FAILS TO CLOSE DUE TO HARDWARE FAULTS),1MSHV3369C---K (HV-3369C FAILS TO CLOSE DUE TO HARDWARE FAULTS),	24 hrs. (24 hrs.)
3.7.4 Atmospheric Relief Valves A. One required ARV line inoperable.	1MSAVPV3371A-D (SG ARV PV3371A FAILS TO OPEN DUE TO RANDOM FAULTS)	30.0 (30.0)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.7.4 Atmospheric Relief Valves B. Two or more required ARV lines inoperable	1MSAVPV3371A-D (SG ARV PV3371A FAILS TO OPEN DUE TO RANDOM FAULTS) AND 1MSAVPV3371B-D (SG ARV PV3371B FAILS TO OPEN DUE TO RANDOM FAULTS)	30.0 (30.0)
3.7.4 Atmospheric Relief Valves C. Three required ARV lines inoperable. (TS LOF)	1MSAVPV3371A-D (SG ARV PV3371A FAILS TO OPEN DUE TO RANDOM FAULTS) AND 1MSAVPV3371B-D (SG ARV PV3371B FAILS TO OPEN DUE TO RANDOM FAULTS)	24 hrs. (24 hrs.)
3.7.5 Auxiliary Feedwater System A. Turbine driven AFW train inoperable due to one inoperable steam supply	1AFXV005B----V (L.O. MANUAL VALVE V005B FAILS CLOSED (IN SEGMENT TI)	30.0 (30.0)
3.7.5 Auxiliary Feedwater System B. One AFW train inoperable for reasons other than Condition A.	1AFPM001B----A (MDP B FAILS TO START DUE TO RANDOM FAULTS)	11.3 (30.0)
3.7.5 Auxiliary Feedwater System B. One AFW train inoperable for reasons other than Condition A.	1AFPT002-----A (TDP P002 FAILS TO START DUE TO RANDOM FAULTS)	24.2 (17.6)
3.7.6 Condensate Storage Tank A. CST Inoperable (TS LOF)	1AFTK-CST-TR-R (CST EXCESSIVE LEAKAGE REQUIRING MAKEUP PRIOR TO 24 HOURS)	24 hrs. (24 hrs.)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.7.6 Condensate Storage Tank A. CST Inoperable (TS LOF)	1AFCV007A----V (CHECK VALVE V007A TRANSFERS CLOSED)	24 hrs. (24 hrs.)
3.7.7 Component Cooling Water A. One CCW train inoperable	1CCPM001C----A (CCW PUMP C FAILS TO START DUE TO RANDOM FAULTS) AND 1CCPM001B----A (CCW PUMP B FTS DUE TO RANDOM FAULTS)	30.0 (30.0)
3.7.7 Component Cooling Water A. One CCW train inoperable	1CCPM001C----A (CCW PUMP C FAILS TO START DUE TO RANDOM FAULTS)	30.0 (30.0)
3.7.7 Component Cooling Water B. Two CCW trains inoperable (TS LOF)	1CCPM001C----A (CCW PUMP C FAILS TO START DUE TO RANDOM FAULTS) AND 1CCPM001B----A (CCW PUMP B FTS DUE TO RANDOM FAULTS)	24 hrs. (24 hrs.)
3.7.8 Service Water System A. One SWS Train inoperable	1SWPM1A------A (SW PUMP 1A RANDOMLY FAILS TO START), 1SWPM1B------A (SW PUMP 1B RANDOMLY FAILS TO START)	26.8 (30.0)
3.7.8 Service Water System B. Two SWS Trains inoperable (TS LOF)	1SWPM1A------A (SW PUMP 1A RANDOMLY FAILS TO START), 1SWPM1B------A (SW PUMP 1B RANDOMLY FAILS TO START)	24 hrs. (24 hrs.)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.7.11 Control Room Air Conditioning System (CRACS) in MODE 1, 2, 3, or 4 E. Two CRACs trains inoperable (TS LOF)	Not Modeled- Documented in PRA basis heat up analysis as not needed to prevent to core damage.	24 hrs. (24 hrs.)
3.7.19 Engineered Safety Features (ESF) Room Coolers A. One required ESF Room Cooler subsystem Train inoperable	1HHMOM001A---X (CHG PMP A FAN COOLER FAILS TO RUN DUE TO RANDOM FAULTS), 1LHMOM003A---X (RHR PUMP 1A FAN COOLER FAILS TO RUN DUE TO RANDOM FAULTS), 1CSMOM002A---X (CS PUMP Q1E13P001A ROOM COOLER RANDOMLY FAILS TO RUN), 1AFMOH005A-TRX (MDAFW PUMP A ROOM COOLER FTR DUE TO RANDOM FAULTS)	9.1 (9.6)
3.7.19 Engineered Safety Features (ESF) Room Coolers A. One required ESF Room Cooler subsystem Train inoperable	1HHMOM001C---X (PMP C FAN COOLER FAILS TO RUN DUETO RANDOM FAULTS) AND 1LHMOM003B---X (RHR PUMP 1B FAN COOLER FAILS TO RUN DUE TO RANDOM FAULTS) AND 1CSMOM002B---X (CS PUMP Q1E13P001B ROOM COOLER RANDOMLY FAILS TO RUN) AND 1AFMOH005B-TRX (MDAFW PUMP B ROOM COOLER FTR DUE TO RANDOM FAULTS)	9.0 (28.4)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.7.19 Engineered Safety Features (ESF) Room Coolers B. Two trains of the same ESF Room Cooler subsystem inoperable	1HHMOM001A---X (CHG PMP A FAN COOLER FAILS TO RUN DUE TO RANDOM FAULTS), 1LHMOM003A---X (RHR PUMP 1A FAN COOLER FAILS TO RUN DUE TO RANDOM FAULTS), 1CSMOM002A---X (CS PUMP Q1E13P001A ROOM COOLER RANDOMLY FAILS TO RUN), 1AFMOH005A-TRX (MDAFW PUMP A ROOM COOLER FTR DUE TO RANDOM FAULTS)	9.1 (9.6)
3.7.19 Engineered Safety Features (ESF) Room Coolers B. Two trains of the same ESF Room Cooler subsystem inoperable	1HHMOM001C---X (PMP C FAN COOLER FAILS TO RUN DUETO RANDOM FAULTS) AND 1LHMOM003B---X (RHR PUMP 1B FAN COOLER FAILS TO RUN DUE TO RANDOM FAULTS) AND 1CSMOM002B---X (CS PUMP Q1E13P001B ROOM COOLER RANDOMLY FAILS TO RUN) AND 1AFMOH005B-TRX (MDAFW PUMP B ROOM COOLER FTR DUE TO RANDOM FAULTS)	9.0 (28.4)
3.8.1 AC Sources – Operating A. One required offsite circuit inoperable	1ACTRSUT1B---F (START UP TRANSFORMER 1B RANDOM FAILURE)	12.7 (30.0)
3.8.1 AC Sources – Operating B. One DG set inoperable	BDGGER43A501AAL (DIESEL 1/2A FAILS TO START ON DEMAND DUE TO RANDOM FAILURE)	30.0 (30.0)
3.8.1 AC Sources – Operating C. Two required offsite circuits inoperable	1ACTRSUT1B---F (START UP TRANSFORMER 1B RANDOM FAILURE),AND 1ACTRSUT1A---F (START UP TRANSFORMER 1A RANDOM FAILURE)	1.4 (6.8)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.8.1 AC Sources – Operating D. One required offsite circuit inoperable. <u>AND</u> One DG set inoperable.	1ACTRSUT1A---F (START UP TRANSFORMER 1A RANDOM FAILURE), BDGGER43A501AAL(DIESEL 1/2A FAILS TO START ON DEMAND DUE TO RANDOM FAILURE), AND BDGGEDIESEL1CAS (DIESEL 1C FAILS TO START ON DEMAND)	0.2 (1.8)
3.8.1 AC Sources – Operating E. Two DG sets inoperable	1DGGER43A502BAL (DIESEL 1B FAILS TO START ON DEMAND DUE TO RANDOM FAILURE), BDGGER43A501AAL(DIESEL 1/2A FAILS TO START ON DEMAND DUE TO RANDOM FAILURE), AND BDGGEDIESEL1CAS(DIESEL 1C FAILS TO START ON DEMAND)	4.3 (27.6)
3.8.1 AC Sources – Operating G. One Automatic Load Sequencer inoperable	1ACARB1G52GX-F (SEQ B1G RELAY 52GX FAILS DUE TO RANDOM CAUSE) & 1ACARB1G4G---F (RANDOM FAILURE OF SEQ B1G RELAY 4G) & 1ACARB1GXG---F (SEQ B1G RELAY XG FAILS DUE TO RANDOM CAUSE) & 1ACCNB1G68G13U (SEQ. B1G AUX. RELAY 68G1 CONTACTS 3,4 SPURIOUSLY OPEN)	17.1 (30.0)
3.8.1 AC Sources H. Three or more required AC Sources inoperable (TS LOF)	1ACTRSUT1B---F ((START UP TRANSFORMER 1B RANDOM FAILURE), and 1ACTRSUT1A---F ((START UP TRANSFORMER 1A RANDOM FAILURE)	24 hrs. (24 hrs.)
3.8.1 AC Sources H. Three or more required AC Sources inoperable (TS LOF)	1ACTRSUT1A---F (START UP TRANSFORMER 1A RANDOM FAILURE) AND BDGGER43A501AAL(DIESEL 1/2A FAILS TO START ON DEMAND DUE TO RANDOM FAILURE), AND BDGGEDIESEL1CAS (DIESEL 1C FAILS TO START ON DEMAND)	0.2 (24 hrs.)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.8.1 AC Sources H. Three or more required AC Sources inoperable (TS LOF)	BDGGER43A501AAL(DIESEL 1/2A FAILS TO START ON DEMAND DUE TO RANDOM FAILURE), AND BDGGEDIESEL1CAS (DIESEL 1C FAILS TO START ON DEMAND), 1DGGER43A502BAL (DIESEL 1B FAILS TO START ON DEMAND DUE TO RANDOM FAILURE)AND BDGGER43A504BAS (DIESEL 2C FAILS TO START ON DEMAND)	0.5 (24 hrs.)
3.8.4 DC Sources – Operating A. One Auxiliary Building DC electrical power subsystem inoperable	1DCBSR42B001AF (RANDOM FAILURE OF DC BUS 1A)	1.4 (1.2)
3.8.4 DC Sources – Operating B. One Auxiliary Building DC electrical power subsystem with battery connection resistance not within limit.	1DCBYR42E002AF (AUXILIARY BUILDING BATTERY 1A FAILS DUE TO RANDOM FAULT)	30.0 (30.0)
3.8.4 DC Sources – Operating D. One required SWIS DC electrical power subsystem battery connection resistance not within limit.	BDCBYR42B523CF (3.8.4 SERVICE WATER BATTERY #3 FAILURE)	30.0 (30.0)
3.8.4 DC Sources – Operating F. Two or more DC electrical subsystems inoperable that result in a loss of function (TS LOF)	1DCBSR42B001AF (RANDOM FAILURE OF DC BUS 1A)	24 hrs. (24 hrs.)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
3.8.7 Inverters – Operating A. One required inverter inoperable	1DCBSR42B001AF (RANDOM FAILURE OF DC BUS 1A)	1.4 (1.2)
3.8.7 Inverters - Operating B. Two or more required inverters inoperable	1ACIVR21E009AF(INVERTER 1A FAILURE),1ACIVE009B-I2F(INVERTER 1B RANDOM FAILURE)	30.0 (30.0)
3.8.9 Distribution Systems Operating D. One or more AC electrical distribution subsystems inoperable for reasons other than Condition A., B, or C	1ACTRSUT1A---F (START UP TRANSFORMER 1A RANDOM FAILURE), BDGGER43A501AAL(DIESEL 1/2A FAILS TO START ON DEMAND DUE TO RANDOM FAILURE), BDGGEDIESEL1CAS (DIESEL 1C FAILS TO START ON DEMAND)	0.2 (1.8)
3.8.9. Distribution Systems Operating E. One or more AC Vital buses inoperable	1ACBSL001A-I2F(VITAL AC PANEL 1A FAILURE), AND 1ACBSL001B-I1F (VITAL AC PANEL 1B FAILURE)	30.0 (30.0)
3.8.9. Distribution Systems Operating F. One Auxiliary Building DC electrical power distribution subsystem inoperable	1DCBSB001ADGSF (125V DC BUS 1A RANDOMLY FAILS (DG START SUPPORT))	14.9 (30.0)
3.8.9. Distribution Systems Operating G. Two trains with inoperable distribution subsystems that result in a loss of safety function (TS LOF)	1ACTRSUT1A---F (START UP TRANSFORMER 1A RANDOM FAILURE), BDGGER43A501AAL(DIESEL 1/2A FAILS TO START ON DEMAND DUE TO RANDOM FAILURE), BDGGEDIESEL1CAS (DIESEL 1C FAILS TO START ON DEMAND)	0.2 (24 hrs)

Table E1.2
Unit 1/Unit 2 TS RICT Estimate Based on CDF(LERF) Limit

TS LCO/Condition	Selected Equipment Description	RICT Calculated for Selected Equipment in Days - CDF(LERF) (Note 1)
Note (1) : RICT are days unless specifically denoted in hours for TS LOF backstop		

2.0 References

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

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

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines"

Enclosure 2

Information Supporting Consistency with Regulatory Guide 1.200, Rev. 2

Table of Contents

1.0 Introduction

This enclosure provides information on the technical adequacy of the Farley Nuclear Plant (FNP) Probabilistic Risk Assessment (PRA) internal events model (including flooding) and the FNP fire PRA model in support of the License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" (Reference 1).

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

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

Section 2 of this enclosure describes requirements related to the scope of the FNP PRA internal events model. Section 3 outlines requirements for the internal events PRA from RG 1.200 and how these are met. Section 4 similarly outlines requirements for the fire PRA from RG 1.200 and how these are met. Section 5 provides general conclusions. Finally, Section 6 lists references used in the development of this enclosure.

2.0 Requirements Related to Scope of FNP PRA Model

The FNP internal events PRA model as referenced in the peer review (Reference 11) is an at-power model (i.e., it directly addresses plant configurations during plant modes 1 and 2 of reactor operation). The model includes both at-power internal events core damage frequency (CDF) and large early release frequency (LERF). Internal flooding is included in both the CDF and LERF models. Note that this portion of the FNP PRA model does not incorporate the risk impacts of external events. The treatment of seismic risk and other external hazards for this application is discussed in Enclosure 3. Various PRA notebooks were used for disposition information contained within Tables E2-2 and E2-4, which are available for inspection.

3.0 Technical Adequacy of FNP Internal Events and Internal Flooding PRA Model

NEI 06-09 requires that the PRA be reviewed to the guidance of Regulatory Guide 1.200, Revision 0 (Reference 5) for a PRA which meets Capability Category (CC) II for the supporting requirements of the American Society of Mechanical Engineers (ASME) internal events at power PRA standard (Reference 6). It also requires that deviations from these capability categories relative to the RICT program be justified and documented. Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09 (Reference 2) takes exception to the reference to RG 1.200, Revision 0, currently listed throughout TR NEI 06-09, Revision 0. The NRC staff requires an assessment of PRA technical adequacy using RG 1.200, Revision 1, and the updated PRA standard which, at the time, was ASME RA-Sb-2005.

The FNP PRA has been subjected to a number of peer reviews and self-assessments, including one performed in accordance with the 2009 version of the PRA Standard (Reference 6) as endorsed with clarifications by RG 1.200, Revision 2 (Reference 3). The FNP PRA Peer review conducted in March 2010 was performed using the process defined in NEI 05-04 (Reference 13) and it was a full-scope peer review. NEI 05-04 (Reference 13) guidance supplants the

NEI 00-02 (Reference 10) guidance for conducting a peer review. The results of the RG 1.200 peer review (Capability Category and Findings) are described in Section 3.1. Section 3.2 summarizes the resolution of findings identified in the RG 1.200 peer review.

The information provided in this section demonstrates that the FNP internal events PRA model (including flooding) meets the requirements of RG 1.200.

3.1 RG 1.200 Peer Review for FNP Internal Events PRA Model against ASME PRA Standard Requirements

The 2009 version of the PRA Standard (Reference 6) contains a total of 326 numbered supporting requirements (SRs) in fourteen technical elements and one configuration control element. Eight of the SRs were determined to be not applicable to the FNP PRA. Thus, a total of 318 SRs are applicable.

Among 318 applicable SRs, 92% met Capability Category II or higher, as shown in Table E2-1.

Table E2-1. Summary of FNP Capability Categories

Capability Category Met	No. of SRs	% of total applicable SRs
CC-I/II/III (or SR Met)	213	66.8%
CC-I	9	2.8%
CC-II	30	9.4%
CC-III	12	3.8%
CC-I/II	13	4.2%
CC-II/III	24	7.6%
SR Not Met	17	5.4%
Total	318	100%

Seventeen SRs were judged to be not met. These were IE-C5, AS-C2, SY-A6, SY-C1, HR-G7, HR-I3, IFEV-B3, IFPP-B2, IFPP-B3, IF-QUA7, IF-SNA4, IFSN-B3, IFSO-B3, QU-A5, QU-C2, QU-F1, and MU-B4. An additional 9 SRs met CC-I, but not CC-II. These were: IE-A5, IE-A9, IE-B3, HR-D2, HR-G1, LE-C2, LE-C9, LE-C11, and LE-C12. The peer review generated 40 Findings. These Findings and their resolutions are described in Section 3.3. These include resolution of the Findings related to the 17 SRs that were not met, and to 5 of the 9 SRs judged to be CC-I. Findings were not issued for the LE SRs judged to be not met, but a discussion of those 4 SRs is also provided in Section 3.2. Thus, the FNP internal events PRA (including flood) satisfies the requirements in NEI 06-09-0-A for PRA quality, consistent with the guidance of RG 1.200.

3.2 Resolution of Findings from RG 1.200 Internal Events Peer Review

Table E2-2 shows the details of the 40 Findings and the associated resolutions developed after the peer review. Resolution of these Findings results in all SRs met to at least Capability Category II. Also included are discussions of the 4 LE SRs judged to be CC-I, but for which no findings were issued.

Table E2-2. Resolution of FNP PRA Peer Review Findings

F&O#	Topic	Finding Description	Status	Disposition
IE-A5-01	PERFORM a systematic evaluation of each system, including support systems, to assess the possibility of an initiating event occurring due to a failure of the system. USE A STRUCTURED APPROACH [SUCH AS A SYSTEM-BY-SYSTEM REVIEW OF INITIATING EVENT POTENTIAL, OR A FAILURE MODES AND EFFECTS ANALYSIS (FMEA), OR OTHER SYSTEMATIC PROCESS] TO ASSESS AND DOCUMENT THE POSSIBILITY OF AN INITIATING EVENT RESULTING FROM INDIVIDUAL SYSTEMS OR TRAIN FAILURES.	There is no evidence of a system by system review of the Farley systems to verify no additional initiators exist. A systematic review of the Farley systems and trains should be performed to ensure that all potential initiators are identified and that the initiators are grouped properly on the basis of impact and frequency. Add a systematic review of the safety and non-safety systems that could cause a plant scram to verify that no additional initiators are needed.	Resolved	This F&O is resolved. A systematic review of the Farley safety and non-safety systems was performed that resulted in the development of a Table C-1 “Farley Initiating Event Identification Analysis” which is documented as part of the Farley Initiating Event Notebook. This table lists each Farley system ordered by a system group identifier, system ID, system description, impact of system loss and treatment of system loss in Farley PRA. The “treatment of system loss” addressed specifically whether the loss of a system would result in an initiating event and how the initiating event was grouped. There is no impact on the PRA model since no additional initiators are identified.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IE-A7-01	In the identification of the initiating events, INCORPORATE (a) events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions), and for which it is determined that the event could also occur during at-power operation (b) events resulting in an unplanned controlled shutdown that includes a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation	Section 2 states that events occurring during Modes 3-6 are considered to determine if they are applicable at-power. Appendix B-1 includes events that occurred at power levels less than 10%. However, the review does not seem to look at the event applicability for Mode 1. Two of the reactor trips at 0% power were due to Source Range Monitors (SRMs). These events would not be applicable to the at-power analysis since the SRM would be replaced by the APRMs for Mode 1. Clarify the review of the LPSD events included in Appendix B-1 and how they are included in the plant specific frequency analysis.	Resolved	This F&O is resolved. These two events were reviewed and it was determined that they should be removed from the plant specific frequency analysis. Appendix B-1 of the Initiating Events Notebook was revised to reflect the changes to the analysis. .

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IE-A9-01	REVIEW plant-specific operating experience for initiating event precursors, for identifying additional initiating events. FOR EXAMPLE, PLANT-SPECIFIC EXPERIENCE WITH INTAKE STRUCTURE CLOGGING MIGHT INDICATE THAT LOSS OF INTAKE STRUCTURES SHOULD BE IDENTIFIED AS A POTENTIAL INITIATING EVENT.	There is no indication that IE precursors such as intake clogging have been performed. Precursor reviews generally include a significant plant event that did not cause a scram but could have if prompt action is not taken. Review significant non-scram events at the plant to determine if any precursors exist.	Resolved	This F&O is resolved. A search was performed using the Condition Reports database for significant non-scram events. A comparison of the results was made to Farley's initiating events list. No new initiating event precursors to plant trips were found. Added methodology and review results in Appendix A of the Initiating Events notebook There is no impact on the PRA model since no additional initiating event precursors are identified.

E-A10-02	For multi-unit sites with shared systems, INCLUDE multi-unit site initiators (e.g., multi-unit LOOP events or total loss of service water) that may impact the model.	The Farley IE notebook indicates that failure of the Service Water (SW) pond dam was included as a special initiator. However, a search of the model did not locate the dam failure. Further, the probability of a loss of the SW pond dam is estimated to be 1.9e-7 failures per year based on the FNP River Water Study (dated 1982). This analysis is based on a generic estimate of 1.9e-5 failures per year for earthen filled dams that in the opinion of Alabama Power Company should be reduced to 1.9e-7 per year due to design, monitoring, maintenance, and responsiveness of the owner to problems. Loss of the dam would result in a dual unit loss of service water. For an event of the magnitude of a dual unit loss of service water, the supporting evidence for reduction of the generic value by a factor of 100 is treated very lightly. An initiating event that would result in a dual event initiator should be included in the initiating event portion of the model. Evidence for reducing the generic dam failure probability is qualitative in nature, and the extension of this information to justify a factor of 100 reduction in the generic probability is not clear and poorly supported. Further, dam failure analysis technology has improved since 1982, and use of the newer approaches to analysis should be considered.	Resolved	A sensitivity analysis was performed to show SW Pond Dam failure's contribution to the CDF and LERF. This F&O is resolved Based on the dam failure assessment study, it was concluded that loss of SW due to a random failure of the dam as an initiating event does not need to be modeled in the internal events PRA based on the screening criteria in IE-C6 (b) of the ASME PRA Standard (Reference 6).

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
		Consider revisiting the estimation of the probability of dam failure using newer technology and better supported calculation. Add the loss of the SW pond dam to the model, if appropriate.		
IE-B1-01	COMBINE initiating events into groups to facilitate definition of accident sequences in the Accident Sequence Analysis and to facilitate quantification.	Events are grouped in general categories. It is not clear that the impact on systems are similar or that the grouped event frequency includes these events Loss of Turbine Building Cooling is grouped with loss of Service Water. However, the frequency for these events is expected to be similar and may have different impact on the PSA systems. Other potential groupings, such as the 7300 bus and 4.16 KV buses identified through the operator interviews were not clearly grouped. In other cases, the review of the events from NUREG/CR-3862 and NUREG/CR-5500 are not directly tied to an initiating event class. Include the impact of the initiator (especially the transient events) on the PSA systems in the model.	Resolved	This F&O is resolved. Table C-1 “Farley Initiating Event Identification Analysis” was created and documented in the Farley Initiating Event Notebook. This table lists each Farley system ordered by a system group identifier, system ID, system description, impact of system loss and treatment of system loss in Farley PRA. The treatment of “system loss” addressed specifically whether the loss of a system would result in an initiating event and how the initiating event was grouped.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IE-C1-01	CALCULATE the initiating event frequency accounting for relevant generic and plant-specific data unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty.	In section 4 of the initiating event notebook, Farley discusses the quantification of the vessel rupture frequency. They present the WASH-1400 median frequency of 1E-07 with the associated error bounds but then proceed to treat that value as a mean. This is mathematically incorrect and introduces a slight non-conservative bias. It is not likely to impact the overall results. Calculate the mean from the median and error factor and use that in the quantification. (Mean should be about 2.7E-07.) There is also a newer generic source that has a better number.	Resolved	This F&O is resolved. Revised section 4.1 of the Initiating Events notebook and added reference 18 (PWROG project: PA-RMSC-0463) to the reference section to include a more current data source.
IE-C5-01	CALCULATE initiating event frequencies on a reactor year basis. INCLUDE in the initiating event analysis the plant availability, such that the frequencies are weighted by the fraction of time the plant is at power.	Farley did calculate their initiating event frequencies on a reactor year basis. However, they did not modify the resultant frequencies to address plant availability. Discussions with the Farley staff indicated that the adjustment was not made as part of quantification either. The frequencies are slightly conservative. The initiating event frequency should be modified to address plant availability. This can be done by multiplying each initiating event frequency by the availability factor or the adjustment can be done as part of the quantification.	Resolved	This F&O is resolved. The adjustment has been made as part of the model quantification. Appendix B-2 of the Initiating Events notebook contains the development of the annual average availability factor.

Table E2-2. Resolution of FNP PRA Peer Review Findings

F&O#	Topic	Finding Description	Status	Disposition
IE-C15-01	CHARACTERIZE the uncertainty in the initiating event frequencies and PROVIDE mean values for use in the quantification of the PRA results.	Table 7 of the Farley Initiating Events Notebook presents the initiating event frequencies for the special initiators but does not characterize the uncertainty. The special initiators are quantified using fault tree analysis so the uncertainty intervals inherently can be quantified based on the uncertainty data for basic events. However, the variance is not presented and there is no discussion of this beyond stating that the frequencies are calculated using fault trees. This is a documentation issue. There is no indication that the uncertainty was not included in the overall model quantification. Document how the uncertainty for the special initiators was characterized/ quantified as part of the discussion in section 3.3 of the Initiating Events Notebook.	Resolved	This F&O is resolved This is a documentation issue: As discussed in the issue statement, the uncertainty of special initiating event is evaluated during quantification process.

Table E2-2. Resolution of FNP PRA Peer Review Findings

F&O#	Topic	Finding Description	Status	Disposition
IE-D1-01	DOCUMENT the initiating event analysis in a manner that facilitates PRA applications, upgrades, and peer review.	Farley did document their initiating event analysis. However, the structure and content of the documentation was such that it was often difficult to trace the identification, grouping and quantification of the IEs in an easy to follow manner. This issue was identified in virtually all Technical Elements of the Farley PRA. It was often difficult to determine what Farley had done to address a given SR and required detailed evaluation of the model and many discussions with the Farley PRA staff. One part of the problem was that in several places, the documentation reflected an earlier version of the model (Version 8 versus Version 9) or did not match the model (treatment of miscalibration errors). This made the PRA difficult to review. However, of greater concern, the documentation could only support applications or updates if a knowledgeable/ experienced engineer was involved. This touches on virtually all PRA documents. 1. Ensure that the documentation reflects the latest version of the model. 2. Review the documentation to see if it has sufficient content and is structured such a less experienced engineer can understand the analysis.	Resolved	This F&O is resolved. Many documents including initiating event notebook and documentation reflecting an earlier version have been updated since the peer review was performed.

Table E2-2. Resolution of FNP PRA Peer Review Findings

F&O#	Topic	Finding Description	Status	Disposition
AS-C2-01	DOCUMENT the processes used to develop accident sequences and treat dependencies in accident sequences, including the inputs, methods, and results.	In the discussion of large, medium, and small Loss Of Coolant Accidents (LOCAs), the operator failure to transfer to low head recirculation is discussed. For large LOCAs, this error is OAR_A_1- -----H, and for other LOCAs (or event trees) the error is OAR_A_2------H. The only difference between the two errors is timing. However, the discussion of OAR_A_2------H indicates that the operator must manually align Component Cooling Water (CCW) cooling to the Residual Heat Removal (RHR) heat exchanger. The discussion of OAR_A_1------H does not include the requirement for the operator to realign CCW to the RHR heat exchanger. The two errors appear to have been modeled correctly, but the difference in the description in the AS notebook is confusing. Add the discussion of the operator realigning CCW to the RHR heat exchanger to the description of OAR_A_1------H.	Resolved	This F&O is resolved. The description for OAR_A_1------H in the Accident Sequence notebook was revised to note that “operator action is still required to align CCW cooling to the RHR heat exchanger.” to be consistent with the description of OAR_A_2------H .

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
AS-C2-02	DOCUMENT the processes used to develop accident sequences and treat dependencies in accident sequences, including the inputs, methods, and results.	Table 2.6-1 of the Farley AS notebook identifies events %LOSSACF and %LOSSACG as Loss of Power to 4kV Bus F and Loss of Power to 4 kV Bus G, respectively. However, the table in Section 2.6.4 identifies these events as Loss of 4160 V Bus F and Loss of 4160 V Bus G, respectively. These two events (Section 2.6.4) are not recoverable by the EDGs because of damage to the respective buses. In Section 2.6.4, the events Loss of Power to 4 kV Bus F and Loss of Power to 4 kV Bus G are labeled as %LOSPF and %LOSPG, respectively. Initiating events %LOSPF and %LOSPG are not included in Table 2.6-1. Table 2.6-1 is incomplete because it is lacking initiating events %LOSPF and %LOSPG. Table 2.6-4 incorrectly characterizes initiating events %LOSSACF and %LOSSACG. Add initiating events %LOSPF and %LOSPG to Table 2.6-1. Correct the descriptions of initiating events %LOSSACF and %LOSSACG in Table 2.6-4.	Resolved	This F&O is resolved. The Accident Sequence notebook was revised to correctly reference the loss of bus initiating events. The descriptions of the %LOSSACF and %LOSSACG events in Section 2.6.4 were not changed because they are correct. Instead, the descriptions for those events were corrected in Table 2.6-1 and events %LOSPF and %LOSPG were added to Table 2.6-1. Documentation was revised.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
SC-A2-01	SPECIFY the plant parameters (e.g., highest node temperature, core collapsed liquid level) and associated acceptance criteria (e.g., temperature limit) to be used in determining core damage. SELECT THESE PARAMETERS SUCH THAT THE DETERMINATION OF CORE DAMAGE IS AS REALISTIC AS PRACTICAL, IN A MANNER CONSISTENT WITH CURRENT BEST PRACTICE. DEFINE COMPUTER CODE-PREDICTED ACCEPTANCE CRITERIA WITH SUFFICIENT MARGIN ON THE CODE-CALCULATED VALUES TO ALLOW FOR LIMITATIONS OF THE CODE, SOPHISTICATION OF THE MODELS, AND UNCERTAINTIES IN THE RESULTS, IN A MANNER CONSISTENT WITH THE REQUIREMENTS SPECIFIED UNDER	The maximum core temperature of two cases of Medium LOCA (CL3-MLO-S2 and CL5-MLO-S1) exceeds 1800F early times after accident, but they are considered as success. In the MAAP analysis notebook describes "only exceeded 1800°F for less than 6 min; considered success." (Appendix B, Table B-1) In addition, there are two SGR cases (S1 and S2) in which the core temperature is oscillating unstably, exceeding 1800 F in some of the later oscillations. It is not clear that these or successes or that a stable configuration has been achieved. It is not clear that the identified cases cannot meet the success criteria for core damage. First possible resolution is to perform analysis using another tool instead of MAAP (e.g., a more detailed model that would allow a higher core damage temperature as a success criterion) for these two cases. Second one is to describe the details in the notebook why the analyst assumes these cases as success.	Resolved	This F&O is resolved. While the core damage criteria of 1800°F was exceeded for a short period of time, these 2 MAAP cases are considered successful pertaining to the core damage success criteria. Attachment 1 has been added to Success Criteria notebook to address the maximum core temperature of two cases of Medium LOCA (CL3-MLO-S2 and CL4-MLO-S1).

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
SC-A5-01	SPECIFY an appropriate mission time for the modeled accident sequences. For sequences in which stable plant conditions have been achieved, USE a minimum mission time of 24 hr. Mission times for individual SSCs that function during the accident sequence may be less than 24 hr, as long as an appropriate set of SSCs and operator actions are modeled to support the full sequence mission time. For sequences in which stable plant conditions would not be achieved by 24 hr using the modeled plant equipment and human actions, PERFORM ADDITIONAL EVALUATION OR MODELING BY USING AN APPROPRIATE TECHNIQUE.	There are two SGR cases, S1 and S2, for which the maximum core temperature is oscillating wildly beyond 24 hours, sometimes exceeding 1800 °F. These cases are evidently considered as successes, though it is not evident that a stable configuration has been reached at 24 or even 30 hours. In addition, there are cases for which the mission time is listed as less than 24 hours without explanation. Either do additional calculations to show the two SGR cases are successes or provide adequate explanation of why they are considered successes and a stable condition has been reached. In addition, provide additional explanation of the mission times that are shorter than 24 hours.	Resolved	This F&O is resolved. MAAP analysis was performed using MAAP 4.0.8 to address two SGR cases. The calculation showed that the maximum core temperature did not oscillate and did not exceed 1800 F for the cases.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
SC-B3-01	When defining success criteria, USE thermal/hydraulic, structural, or other analyses/evaluations appropriate to the event being analyzed, and accounting for a level of detail consistent with the initiating event grouping and accident sequence modeling.	The current success criteria for LOCAs are based on plant capabilities and system responses. Although the definitions for small, medium and large break LOCAs are reasonable based on this criteria, the specific break sizes associated with the transitions between the LOCA definitions have not been adequately justified. Currently the break sizes are based on the original IPE criteria and no thermal hydraulic analyses of the break sizes have been performed. Per the requirement, thermal hydraulic evaluations are required at a level of detail to support the definitions/break sizes so that the appropriate initiating event frequencies can be determined. Several utilities' PRAs were dramatically impacted when the MAAP code was used to determine actual break sizes and some utilities determined that an additional fourth size LOCA was required to adequately model their plant. This has the potential to dramatically impact the CDF.	Resolved	This F&O is resolved. MAAP analyses were performed for a 6" break LOCA which is a lower end of large LOCA spectrum and upper end of the medium LOCA spectrum. The MAAP analyses shows that the LOCA is able to be mitigated by either medium LOCA success criteria or large LOCA success criteria. .

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
SC-B3-01 (continued)	See above	**Supplemental Comments:** This comment is a general comment on thermal-hydraulic analysis for Farley. More plant-specific analysis would be required. According to your notebook, break sizes for MAAP analysis are as follows: - Large LOCA : 8.25 ft^2 (about 39 in diameter) - Medium LOCA : 2.18E-02 ft^2, 4.91E-02ft^2, 1.36E-01ft^2 (2 in, 3in, 5 in diameter) - Small LOCA : 7.64E-04ft^2, 5.45E-03ft^2, 2.18E-02ft^2 (0.37 in, 1 in, 2 in diameter) The above break sizes are different from NUREG/CR-6928 (Reference 14). Furthermore, they do not appear to explicitly cover the full range of potential LOCAs (from 5 inches up to 39 inches does not appear to be explicitly addressed).	See above	See above

SC-B3-01 (continued)	See above	According to NUREG/CR-6928 (Reference 14), the break sizes for LOCA are defined as follows: - LLOCA : greater than 6 inches inside diameter (D.2.2) ---> about 0.2 ft^2 - MLOCA : between 2 and 6 inches inside diameter (D.2.4) -- -> about 0.02 ft^2 ~ 0.2 ft^2 - SLOCA : between 0.5 and 2 inches inside diameter (D.2.19) ---> about 0.00005 ft^2 ~ 0.02 ft^2 The success criteria change for the different break classes but there is no analysis to show that the success criteria are appropriate for both the upper and lower end of the break spectrums. For example, the primary difference between LLOCA and MLOCA is typically the number of accumulators required and possibly the number of pumps required. The primary difference between MLOCAs and SLOCAs is that secondary side heat removal is needed for small LOCAs. However, the MAAP analyses do not show that for LOCAs greater than 2 inches, the break is sufficient to remove decay heat while below 2 inches secondary heat removal is required. More and appropriate selection of break size would be required, such as 6 inches, 0.5 inches, etc. Develop LOCA break sizes based	See above	See above

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
		on Farley specific flow capacities and required systems.		
SC-B5-01	CHECK the reasonableness and acceptability of the results of the thermal/hydraulic, structural, or other supporting engineering bases used to support the success criteria.	This SR requires that the reasonableness and acceptability of the SC results be verified. Although there was a table added to the Success Criteria (SC) notebook (during the last few days prior to the peer review) that compares the SCs for Farley to SCs for Summer and Turkey Point, there was no text discussing the table, how the comparison was done, and the reasonableness/ acceptability of any differences between Farley and either Summer or Turkey Point. This is a documentation issue rather than a technical issue since the comparison was apparently done. However, there is no basis in the documentation to determine whether the work to actually verify the reasonableness of the SCs was completed in accordance with the intent of the standard. Add discussion to the system notebook that references Table B and, at least at a high level, explains how the comparison was done and what was done if differences were found. At least, provide a couple of examples to illustrate this process.	Resolved	This F&O is resolved. New text concerning the reasonableness of the SCs has been incorporated into Sections 3.0 of Success Criteria notebook.

SY-A8-01	ESTABLISH the boundaries of the components required for system operation. MATCH the definitions used to establish the component failure data.	In the diesel generator model, the diesel generator, the output breakers, the fuel oil transfer pumps, the sequence relays and the Local Control Panel are all modeled individually. However, Farley uses NUREG/CR-6928 (Reference 14) as the source of their generic diesel generator data and collects plant specific data in accordance with the 6928 component boundaries. The NUREC/CR-6928 (Reference 14) diesel generator component boundary explicitly includes the output breaker and the fuel oil system (without much definition) Thus, the component boundaries as used in the Farley diesel generator system model do not match the component boundaries used for collecting the failure data. Furthermore, the component boundaries used to derive the generic common cause boundaries do not match the component boundaries used to develop the generic failure rates. For the most part, Farley has made the appropriate adjustments to match the two divergent data sets. However, the generic common cause data for diesel generators had an event whose description was such that it could be interpreted as either involving fuel oil transfer pumps or not. The decision was made to include the event as a diesel failure because it would be conservative. The component boundary definitions in the Systems and Data Analysis	Resolved	This F&O is resolved. The modeling approach is valid because: i) The modeled fuel oil transfer pumps are external fuel oil pumps to makeup day tanks which are not sufficient to supply fuel oil to DGs for 24 hours mission time. The pumps are required to makeup fuel oil from storage tank. ii) DG Output circuit breaker, sequence relays and the Local Control Panel are modeled separately because thee of five Farley DGs are shared by two units. Even though explicit modeling of the circuit breaker is somewhat conservative, the proper dependency model is reflected in the model.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
		Notebooks were not very detailed so this was difficult to identify. Farley needs to adjust their data collection and quantification to collect and quantify the diesel generator system failure data consistent with how the system is modeled. Farley also needs to review their component boundary definitions to ensure that they are sufficiently detailed to identify exactly what is included within each component and that are consistent from the model to the system notebooks to the data analysis notebook to the common cause failure analysis.		
SY-A9-01	If a system model is developed in which a single failure of a super component (or module) is used to represent the collective impact of failures of several components, PERFORM the modularization process in a manner that avoids grouping events with different recovery potential, events that are required by other systems, or events that have probabilities that are dependent on the scenario.	The system model boundary is not clearly defined between the notebook and the model. Example is room cooling for Emergency Core Cooling System (ECCS) system is model as part of the system but is listed as a dependent system to the ECCS. AFW discussion of boundary includes condensate tank and steam supply up to steam generators, but later in the notebook defines condensate and steam supply as support systems. See also SY-A8-01 for diesel boundary issues.	Resolved	This F&O is resolved. The system notebooks were reviewed and modified as needed to reflect the boundary of the system as shown in the model. The support system sections were reviewed and corrected as needed to reflect the support systems as modeled. Farley PRA System Analysis Notebooks.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
SY-A23-01	DEVELOP system model nomenclature in a consistent manner to allow model manipulation and to represent the same designator when a component failure mode is used in multiple systems or trains.	The system model nomenclature did not consistently use the fault tree guideline definitions in the naming convention. Examples include: guide has FW as feedwater system but model uses MF as system designator, RF component type identifier is not match the guide, room coolers are modeled with the system supporting. The room cooler system designator is the same as the ECCS pump. Farley should review their naming convention and make sure it is applied consistently in all models.	Resolved	This F&O is resolved. The naming conventions have been updated in the Farley Fault Tree Analysis Guidelines notebook. Specifically the identifier FW was changed to MF for Main Feedwater and the component description for identifier RF was changed to Refrigeration Unit..

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
SY-B6-01	PERFORM engineering analyses to determine the need for support systems that are plant-specific and reflect the variability in the conditions present during the postulated accidents for which the system is required to function.	The room heatup calculations for the Engineered Safety Features (ESF) pump rooms and ESF electrical equipment rooms are excellent. But some calculation results are mismatched with documents and references, the others are conservatively applied into fault tree model. Description of Ref.12 and HVAC system notebook are mismatched with Ref.4. The calculations results show the temperature of the ESF equipment rooms during 30 days after loss of Heating, Ventilation and Air Conditioning (HVAC) condition. Some document errors occurred using 30-day calc. results. If the calc. results for ESF pump rooms and electrical equipment rooms would be checked for 24 hours, temperature of some rooms will be lower than the limit. If then, the system fault trees does not develop "room cooling failure" any more for those cases. Descriptions should be matched.	Resolved	This F&O is resolved. This is a documentation issue. The references were corrected. The model was checked for conservative room cooler failure modeling as a result of interpretation of the calculation results. The HVAC model for the Engineered Safety Features (ESF) pump rooms and ESF electrical equipment rooms were updated based on up-to-date room heatup calculations. Farley PRA System Analysis Notebooks.
SY-C1-01	DOCUMENT the systems analysis in a manner that facilitates PRA applications, upgrades, and peer review.	In section 6.1.7 of the system notebooks for AFW, CCW, Containment Cooling, Containment isolation, Containment Spray, ECCS, IA, MS, SW incorrect reference information to test and maintenance is provided. Farley needs to correct the references for test and maintenance information.	Resolved	This F&O is resolved. This is a documentation issue. The references were corrected. Farley PRA System Analysis Notebooks.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
HR-D2-01	FOR SIGNIFICANT HFES, USE DETAILED ASSESSMENTS in the quantification of pre-initiator HEPs. USE SCREENING VALUES BASED ON A SIMPLE MODEL, SUCH AS ASEP IN THE QUANTIFICATION OF THE PREINITIATOR HEPS FOR NONSIGNIFICANT HUMAN FAILURE BASIC EVENTS. When bounding values are used, ENSURE they are based on limiting cases from models such as ASEP.	Farley develops detailed restoration errors for three events and applies this probability to most of the remaining events without any specific evidence through procedures or tests that the events are similar enough that the same values should apply. The values for these restoration errors could be significantly over-estimated since the value applied is not shown to be directly applicable to the event analyzed. Detailed analysis should only be applied to the event analyzed or to directly applicable events where procedures and actions are similar (SW pump trains with identical restoration type errors through similar procedures). Perform detailed analysis on all events to verify the applicability used or use screening values for those events not explicitly analyzed with a detailed analysis.	Resolved	This F&O is resolved. A revision to Table 8-2 of the HRA notebook has been incorporated providing a more detailed explanation of the approach used. The pre-initiator approach relies on detailed THERP assessments that are mapped to similar HFEs.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
HR-D2-02	FOR SIGNIFICANT HFES, USE DETAILED ASSESSMENTS in the quantification of pre-initiator HEPs. USE SCREENING VALUES BASED ON A SIMPLE MODEL, SUCH AS ASEP IN THE QUANTIFICATION OF THE PREINITIATOR HEPS FOR NONSIGNIFICANT HUMAN FAILURE BASIC EVENTS. When bounding values are used, ENSURE they are based on limiting cases from models such as ASEP.	The screening probability used for unanalyzed events is 1E-4. This is significantly lower than the base screening HEP from ASEP which is median failure rate of 3E-2. Even if credit is taken for a recovery factor such as post-maintenance testing or independent verification, then the screening value would be approximately 8E-3. The screening values used are significantly below the screening values recommended in Technique for Human Error Rate Prediction (THERP) and Accident Sequence Evaluation Program (ASEP). Review the Pre-accident HRA screening values that are used and be consistent with ASEP as discussed in the SR.	Resolved	This F&O is resolved. A revision to Table 8-2 of the HRA notebook has been incorporated providing a more detailed explanation of the approach used. The pre-initiator approach relies on detailed THERP assessments that are mapped to similar HFEs.
HR-G1-01	PERFORM DETAILED ANALYSES FOR THE ESTIMATION OF HEPS FOR SIGNIFICANT HFES. USE SCREENING VALUES FOR HEPS FOR NONSIGNIFICANT HUMAN FAILURE BASIC EVENTS.	The top HRA events in the QU notebook are not developed in the HRA notebook. Example 1RTOPMANRTNSGH and OMG_A_2-------H. These events appear in several of the top 50 cutsets and are thus significant to the risk assessment Develop HRAs for these events and include in the HRA calculation.	Resolved	This F&O is resolved. Included the events in the HRA calculator (section 10.92 and 10.93) file using the values found in NUREG CR-5500 and WCAP-15831.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
HR-G7-01	For multiple human actions in the same accident sequence or cut set, identified in accordance with supporting requirement QU-C1, ASSESS the degree of dependence, and calculate a joint human error probability that reflects the dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including (a) time required to complete all actions in relation to the time available to perform the actions (b) factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.) (c) availability of resources (e.g., personnel)	The top HRA cutset combinations in the QU notebook are not addressed in the HRA dependency analysis. These events appear in several of the top 50 cutsets and are thus significant to the risk assessment Explicitly evaluate the top HRA combinations in the dependency analysis.	Resolved	This F&O is resolved. An HRA Dependency Analysis was conducted and incorporated into the Revision 9 model quantification. This analysis has been incorporated into the HRA notebook as Attachment C.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
HR-G7-02	For multiple human actions in the same accident sequence or cut set, identified in accordance with supporting requirement QU-C1, ASSESS the degree of dependence, and calculate a joint human error probability that reflects the dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including (a) time required to complete all actions in relation to the time available to perform the actions (b) factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.) (c) availability of resources (e.g., personnel)	Attachment C to the HRA notebook performs the dependency assessment, but the dependency factors are based upon 2004 HRA values. The multiplication factors in the rule file are to be based upon current HRA. The recovery rules seem to address dependence with factors greater than one and only then for 5 events. This is not consistent with the dependence methods. Update the HRA dependence evaluation to be consistent with industry practices.	Resolved	This F&O is resolved. An HRA Dependency Analysis was conducted and incorporated into the Revision 9 model quantification. This analysis has been incorporated into the HRA notebook as Attachment C.

Table E2-2. Resolution of FNP PRA Peer Review Findings

F&O#	Topic	Finding Description	Status	Disposition
HR-I3-01	DOCUMENT the sources of model uncertainty and related assumptions associated with the human reliability analysis.	Assumptions are listed in the individual HRA analyses. However, some major assumptions normally associated in an HRA analysis, such as default minimum values for pre- and post-accident HRAs, are not included in the analysis. In addition, uncertainty based on using the same HRA probability for all manual valve misalignments is ripe for an uncertainty evaluation. Also the HRA calculation does not address the different types of uncertainty that is included in other Farley document packages. Review the EPRI report on HRA uncertainties and see if any will apply to Farley. Documentation of sources of uncertainty is required by the SR Include a source of uncertainty in the HRA calculation.	Resolved	This F&O is resolved. A document was created to address HRA Uncertainty for the Farley Revision 9 model. It can be found as Attachment F in the HRA notebook.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
DA-C14-01	EXAMINE coincident unavailability due to maintenance for redundant equipment (both intrasys-tem and intersystem) THAT IS A RESULT OF A PLANNED, REPETITIVE ACTIVITY based on actual plant experience. CALCULATE coincident maintenance unavailabilities that are a result of a planned, repetitive activity that reflect actual plant experience. Such coincident maintenance unavailability can arise, for example, for plant systems that have installed spares (i.e., plant systems that have more redundancy than is addressed by tech specs).	Several of the data sets used in the Farley database are based on information that is getting dated. The period over which these data were collected is 1984 through 2001, or earlier. The affected data sets include Table 4 (simultaneous maintenance on redundant equipment), offsite power recovery, and plant-specific data used for failure rates, probabilities, and unavailability. For RIR application, periodically updated plant specific data is required. These data sets need to be updated using more recent information.	Resolved	This F&O is resolved. The data were updated using more recent industry generic data and plant specific experience data.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IFPP-B2-02	DOCUMENT the process used to identify flood areas. For example, this documentation typically includes (a) flood areas used in the analysis and the reason for eliminating areas from further analysis (b) any walkdowns performed in support of the plant partitioning	The IF notebook provides descriptions about flood areas within four (4) buildings, such as auxiliary building, diesel building, service water intake structure, and turbine building. There is no description about the other buildings. Even though they are not risk-significant, the description about the reason why those buildings are not analyzed is needed. The screened/ eliminated areas are not considered in the analysis. Possible resolution is to add information about the screened/eliminated areas and buildings in terms of internal flooding analysis.	Resolved	This F&O is resolved. New text concerning screened/eliminated areas and buildings has been incorporated into the Section 3.1 of the Flooding notebook.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IFPP-B2-03	DOCUMENT the process used to identify flood areas. For example, this documentation typically includes (a) flood areas used in the analysis and the reason for eliminating areas from further analysis (b) any walkdowns performed in support of the plant partitioning	Even though Farley has areas that are common between both units the documentation of how multi-unit impacts were addressed could not be located. Discussions with the Farley PRA staff did reveal that Farley had considered multi-unit effects, However, the documentation of how Farley explicitly considered the potential for multi-unit floods is not well presented. The IF Notebook needs to be updated to address the potential for multi-unit floods or the propagation of a flood in one unit to the other unit via shared spaces. Farley needs to explicitly describe how they dealt with the evaluation multiunit effects for areas where there shared spaces. The basis for any screening of such areas should be explicitly described in the text as well as in the screening table.	Resolved	This F&O is resolved. New text concerning multi-unit impacts has been incorporated into the Section 3 and 12.5 of the Flooding notebook.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IFSN-A2-01	For each defined flood area and each flood source, IDENTIFY plant design features that have the ability to terminate or contain the flood propagation. INCLUDE the presence of (a) flood alarms (b) flood dikes, curbs, sumps (i.e., physical structures that allow for the accumulation and retention of water) (c) drains (i.e., physical structures that can function as drains) (d) sump pumps, spray shields, water-tight doors (e) blowout panels or dampers with automatic or manual operation capability	The flood analysis does discuss the potential effect of: alarms, structure such as curbs and sumps, drains, sump pumps, watertight doors; However any direct application of these factors was hard to find. The factors most often explicitly credited was the credit for jacketed piping eliminating spray considerations and air/water tight doors stopping propagation.. The remarks column in table 7-1 does seem to reference hatches as propagation paths but it is not clear that impact of drains and curbs or the like were considered for propagation. Farley should update the flood documentation to provide more information on plant features that can impact the propagation or retention for each flood scenario, especially anywhere that non-watertight doors, berms or curbs are credited	Resolved	This F&O is resolved. New text has been incorporated in Table 6-1 through 6-4, Tables 7-1 through 7-4, Table 9-1, Section 12.3 of the Flooding notebook.
IFSN-A4-01	ESTIMATE the capacity of the drains and the amount of water retained by sumps, berms, dikes, and curbs. ACCOUNT for these factors in estimating flood volumes and SSC impacts from flooding.	In the IF Notebook, there was extensive discussion with respect to treatment of drains, there was explicit evidence that drains were considered as propagation paths for several flood scenarios. However, no explicit estimation of drain capacities could be found. This is a direct violation of SR. Farley should consider adding a table that explicitly includes drain capacities.	Resolved	This F&O is resolved. New text has been incorporated into the appropriate sections of the Internal Flooding Analysis Notebook.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IFSN-B3-01	DOCUMENT sources of model uncertainty and related assumptions associated with the internal flood scenarios.	The IF Notebook did not seem to include assumptions related to the flood scenario selection in a coherent fashion nor did there seem to be any discussion concerning sources of uncertainty. Farley needs to include a section in the IF Notebook to discuss the IF assumptions and sources of uncertainty. A section on assumptions could be included in each section (such as was done for section 9 and 11) or a single section encompassing all tasks could be added	Resolved	This F&O is resolved. New text concerning uncertainty and assumptions has been incorporated into the appropriate sections of the Internal Flooding Analysis Notebook.
IFEV-B3-01	Document sources of model uncertainty and related assumptions associated with the internal flood-induced initiating events.	The Farley PRA flooding analysis indicates that sources of uncertainty were not documented because of the low contribution to CDF and LERF from flooding. Although this is true, this SR requires that sources of model uncertainty and related assumptions associated with the internal flood-induced initiating events be documented. Include a discussion of uncertainty and assumptions related to internal flood initiating events. This finding is related to other internal flooding SRs that discuss documentation of uncertainty.	Resolved	This F&O is resolved. New text concerning uncertainty and assumptions has been incorporated into the appropriate sections of the Internal Flooding Analysis Notebook.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IFQU-A6-01	For all human failure events in the internal flood scenarios, INCLUDE the following scenario-specific impacts on PSFs for control room and ex-control room actions as appropriate to the HRA methodology being used: (a) additional workload and stress (above that for similar sequences not caused by internal floods) (b) cue availability (c) effect of flood on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) (d) flooding-specific job aids and training (e.g., procedures, training exercises)	HRA for flooding event was performed but the base is different from internal event HRA. It seems that there is version mismatch. Possible resolution is to update the HRA for flooding events like HRA for internal events.	Resolved	This F&O is resolved. A plant-specific calculation describes the HRA methodology for flooding PRA and flooding human failure events were added to the HRA Calculator database.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IFQU-A7-01	PERFORM internal flood sequence quantification in accordance with the applicable requirements described in the quantification (QU).	Quantification of flooding event does not perform uncertainty analysis and dependency analysis. Section 10.1.7 explains the dependencies between human interactions, and Farley performed dependency analysis when quantifying the flood CDF. However, there is no description of calculation results about the dependencies. Technical Items are missing. Possible resolution is to perform and provide uncertainty analysis and dependency analysis, even though the flood risk is not significant.	Resolved	This F&O is resolved. An HRA Dependency Analysis was conducted and incorporated into the Revision 9 model quantification. This analysis has been incorporated into the HRA notebook as Appendix C.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IFQU-A11-01	CONDUCT walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify inputs to (a) engineering analyses (b) human reliability analyses (c) spray or other applicable impact assessments (d) screening decisions	Internal Flooding Analysis Notebook Appendix .A does not describe the information related with human reliability analyses and screening decisions. Possible resolution is to provide the room for 1) Operator mitigation action and 2) Reason of screening decisions. **Supplemental Comments:** According to ASME Standard, IFQU-A11, human actions (and human reliability analysis) modeled for each flood area's quantification are verified via flood walk downs. Also, the reason of screening decision should be verified via walk downs. Proposed resolution : The walk down sheet for each flood area add two more sections as follows: G. related human actions H. Screening Decision In case of screening, table 6-1 in the notebook would be a good reference. In case of HRA, table 10-2 and 10-3 would be a good reference.	Resolved	This F&O is resolved. Although qualitative screening is documented in Table 6-1 to 6-4 of the Internal Flooding notebook, the tables did not include any human actions. Section 6 of the Internal Flooding notebook lists the screening criteria which includes human mitigating actions as a criterion (criterion d.). However most of the flood locations in Tables 6-1 to 6-4 were qualitatively screened based on criterion a or b. None were screened on criterion d, which shows that although human actions were considered as a screening criterion, there were no applicable areas in the Farley flooding PRA. .

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
IFQU-B3-01	DOCUMENT sources of model uncertainty and related assumptions associated with the internal flood accident sequences and quantification.	The Farley PRA flooding analysis indicates that sources of uncertainty were not documented because of the low contribution to CDF and LERF from flooding. Although this is true, this SR requires that sources of model uncertainty and related assumptions associated with the internal flood-induced initiating events be documented. Include a discussion of uncertainty and assumptions related to internal flood initiating events. This finding is related to other internal flooding SR that discusses documentation of uncertainty.	Resolved	This F&O is resolved. New text concerning uncertainty and assumptions has been incorporated into the appropriate sections of the Internal Flooding Analysis Notebook.
QU-F1-01	DOCUMENT the model quantification in a manner that facilitates PRA applications, upgrades, and peer review.	The maintenance related mutually exclusive events are stated to be based on Tech Spec disallowed maintenance conditions. The mutually exclusive logic was based upon FNP-0-ACP-52.1 but was not referenced as the source of the mutually exclusive logic. The review of the QU notebook referenced the incorrect document the development of the mutually exclusive logic. Update the documentation to reflect the actual references.	Resolved	This F&O is resolved. Documentation reference has been updated in the Internal PRA Quantification Notebook.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
QU-F4-01	DOCUMENT the characterization of the sources of model uncertainty and related assumptions.	The information regarding the assumptions and sources of uncertainty is located in Appendix D of the Farley QU notebook. However, this appendix is not referenced in the QU notebook, neither is it included in the notebook's table of contents. The only reference to the appendix is in the Revision 9 Roadmap and Quality Self-Assessment document, and in this document it is misidentified as Appendix A. References to this document are either nonexistent or incorrect. Even though the document contains a lot of good information, it is almost impossible to locate. Correct the QU notebook table of contents to include Appendix D and its title. Add information to Section 2 that references the appendix. Correct the reference to the appendix in the Revision 9 Roadmap.	Resolved	This F&O is resolved. Added Appendix D to the Internal PRA Quantification Notebook. Corrected references to the Appendix in the Revision 9 Roadmap.

Table E2-2. Resolution of FNP PRA Peer Review Findings

F&O#	Topic	Finding Description	Status	Disposition
N/A (LE-C2)	INCLUDE realistic treatment of feasible operator actions following the onset of core damage consistent with applicable procedures, e.g., EOPs/SAMGs, proceduralized actions, or Technical Support Center guidance.	The Farley PRA LERF model relies largely on human error probabilities taken from the WCAP-16341-P. Because the WCAP HEPs are generic rather than plant-specific, they were derived as conservative estimates.	No Finding	Although this SR was determined to be CC-I by the peer review, no F&O was made. The conservatism introduced by meeting this SR at CC-I level would not significantly affect the conclusions made based on the Farley PRA results, including the calculation of RICT values. The major contributors to Farley LERF (97% of total LERF) are containment bypass scenarios such as interfacing systems LOCA and containment isolation failure concurrent with core damage scenarios. For such LERF scenarios with containment bypassed or containment isolation failed, operator actions which can be credited in reducing the likelihood of early containment failure after core damage are of little importance because containment barrier is already failed at the time of the core damage.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
N/A (LE-C9)	JUSTIFY any credit given for equipment survivability or human actions under adverse environments.	No credit is taken for either equipment operation or human actions in adverse environments.	No Finding	Although this SR was determined to be CC-I by the peer review, no F&O was made. The conservatism introduced by meeting this SR at CC-I level would not significantly affect the conclusions made based on the Farley PRA results, including the calculation of RICT values. Major contributors to Farley LERF (97% of total LERF) are containment bypass scenarios such as interfacing systems LOCA and containment isolation failure concurrent with core damage scenarios. For such LERF scenarios with containment bypassed or containment isolation failed, operator actions or mitigation systems which can be credited in reducing the likelihood of early containment failure after core damage are of little importance because containment barrier is already failed at the time of the core damage.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
N/A (LE-C11)	JUSTIFY any credit given for equipment survivability or human actions that could be impacted by containment failure.	No credit was taken in the Farley PRA for equipment or operator actions impacted by containment failure. The WCAP-16341-P methodology conservatively does not credit containment sprays for fission product scrubbing or pressure suppression for the containment failure.	No Finding	Although this SR was determined to be CC-I by the peer review, no F&O was made. The conservatism introduced by meeting this SR at CC-I level would not significantly affect the conclusions made based on the Farley PRA results, including the calculation of RICT values. Major contributors to Farley LERF (97% of total LERF) are containment bypass scenarios such as interfacing systems LOCA and containment isolation failure concurrent with core damage scenarios. For such LERF scenarios with containment bypassed or containment isolation failed, operator actions or mitigation systems which can be credited in reducing the likelihood of early containment failure after core damage are of little importance because containment barrier is already failed at the time of the core damage.

Table E2-2. Resolution of FNP PRA Peer Review Findings				
F&O#	**Topic**	**Finding Description**	**Status**	**Disposition**
N/A (LE-C12)	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 conservative and realistic treatment for non-significant accident progression sequences.	The LERF frequency calculated in the Farley PRA is so low that no review was performed to reduce LERF based on engineering analysis to support equipment operation or operator action after containment failure.	No Finding	Although this SR was determined to be CC-I by the peer review, no F&O was made. The conservatism introduced by meeting this SR at CC-I level would not significantly affect the conclusions made based on the Farley PRA results, including the calculation of RICT values. Major contributors to Farley LERF (97% of total LERF) are containment bypass scenarios such as interfacing systems LOCA and containment isolation failure concurrent with core damage scenarios. For such LERF scenarios with containment bypassed or containment isolation failed, operator actions or mitigation systems which can be credited in reducing the likelihood of early containment failure after core damage are of little importance because containment barrier is already failed at the time of the core damage.

Table E2-2. Resolution of FNP PRA Peer Review Findings

F&O#	Topic	Finding Description	Status	Disposition
MU-B4-01	PRA Upgrades shall receive a peer review and the peer review section of each respective part of the standard for those aspects of the PRA that have been upgraded.	There is no reference to a peer review for upgrades. Did not find a section which addressed upgrades (not updates) to the PRA specifically involving changes to key PRA software. This is a direct violation of an SR. Revise either NL-PRA-001 or NL-PRA-002 to explicitly require a peer review for PRA upgrades (i.e. methodology change or major software change etc.)	Resolved	This F&O is resolved. Relevant SNC Procedure was revised to require a peer review following an upgrade of the PRA model.

4.0 Technical Adequacy of FNP Fire PRA Model

NEI 06-09 requires that the PRA be reviewed to the guidance of Regulatory Guide 1.200, Rev 2 (Reference 3) for a PRA which meets Capability Category (CC) II for the supporting requirements of the American Society of Mechanical Engineers (ASME) fire events at power PRA standard. It also requires that deviations from these capability categories relative to the RICT program be justified and documented.

The FNP Fire PRA has undergone a RG 1.200, Revision 2 (Reference 3) Peer Review against the ASME PRA Supporting Requirements (SRs) by a team of knowledgeable industry (vendor and utility) personnel. The review (Reference 12) was conducted by the Westinghouse Owners Group in accordance with NEI 07-12 as endorsed by RG 1.200 Rev 2 (Reference 3). The conclusion of the review was that the FNP methodologies being used were appropriate and sufficient to satisfy the ASME/ANS PRA Standard RA-Sa-2009 (Reference 6). The review team also noted that NUREG/CR-6850 methodologies were applied correctly.

The summary of the peer review findings exhibited the following statistics for the evaluation of elements to the combined PRA Standard. For the FNP Fire PRA, 88% of the SRs were assessed at Capability Category II or higher, including 8% of the SRs being assessed at Capability Category III. The FNP Fire PRA had an additional 5% of the applicable SRs assessed at the Capability Category I level. The Fire PRA was found to not meet 7% of the applicable SRs.

The Westinghouse Peer Group concluded that the Farley Fire PRA is consistent with the ASME/ANS PRA Standard and supports risk-informed applications. As a result of the peer review and the fire risk evaluation process the FNP Fire PRA has undergone additional model refinements. These refinements were made consistent with the methodologies that were reviewed during the FNP Peer Review.

This enclosure provides a detailed assessment of each of the findings identified by the Peer Review team.

4.1 RG 1.200 Peer Review for FNP Fire PRA Model against ASME PRA Standard Requirements

The ASME/ANS RA-SA-2009 version of the PRA Standard (Reference 6) contains a total of 173 numbered supporting requirements (SRs) in 13 technical elements. The configuration control element has 10 additional SRs. Thus, a total of 183 SRs were assessed.

Among 183 SRs, 29 were determined to be not applicable to the FNP Fire PRA either due to the fact that the requirements were not applicable to the FNP approach or the technical element was not used for the FNP analysis (i.e., QLS and QNS). Of the 154 total applicable SRs, approximately 87% met Capability Category II or higher, as shown in Table E2-3.

Table E2-3. Summary of FNP Fire Events Capability Categories

Capability Category Met	No. of SRs	% of total SRs	% of total applicable SRs
Met	100	54.6%	64.9%
Not Met	13	7.2%	8.4%

Table E2-3. Summary of FNP Fire Events Capability Categories			
Capability Category Met	**No. of SRs**	**% of total SRs**	**% of total applicable SRs**
CC I	9	4.9%	5.8%
CC II	5	2.7%	3.3%
CC III	14	7.6%	9.2%
CC I/II	4	2.3%	2.6%
CC II/III	9	4.9%	5.8%
NA	29	15.8%	-
NR	0	-	-
Total	183	100.0%	100%

The peer review generated 31 Findings out of which 13 SRs were judged to be not met. These were PP-B2, PP-B3, PP-C3, PRM-B2, FSS-D7, FSS-D8, FSS-D11, FSS-F1, FQ-C1, FQ-D1, FQ-E1, FQ-F1, and UNC-A1. An additional 9 SRs met CC-I, but not CC-II. These were: CS-B1, FSS-B2, FSS-C1, FSS-C2, FSS-D3, FSS-E3, FSS-F2, FSS-G6, and FSS-H5. The Findings and resolutions associated with these SRs are described in Section 4.3. Thus, the FNP fire PRA satisfies the requirements in NEI 06-09-0-A for PRA quality, consistent with the guidance of RG 1.200.

4.2 Resolution of Findings from RG 1.200 Fire PRA Peer Review

Table E2-4 shows the details of the 31 Findings and the associated resolutions developed after the peer review.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
CS-B1-01	The Farley breaker Coordination documentation was identified to be incomplete based on the Farley Fire PRA components being credited.	The PRA components are not explicitly discussed in a coordination calculation. Calculation SE-C051326701-002 is titled NSCA components; however, informal review has determined that PRA components are addressed. The information to determine the status of coordination for PRA components consists of informal queries and spreadsheets. Supporting Requirement CS-B1-01 Category II requires all buses credited in the Fire PRA to be analyzed for proper over current coordination and protection. Revise calculation SE-C051326701-002 to formally validate that PRA buses are addressed for proper coordination and incorporate results into the Fire PRA model as needed.	Resolved	This F&O is resolved. The Farley circuit analysis calculation, SE-C051326701-002, has been updated to address all coordination concerns. This update identified two panels that were found to not be coordinated; all other panels were dispositioned as acceptable. The two panels are N1R19L00504 and N2R19L00504. Calculation PRABC-F-11-003 (Cable Selection and Detailed Circuit Analysis) has been updated to address this coordination issue. Based on these conclusions these two panels have been failed in every scenario for the Farley Fire PRA. Associated Circuits Analysis Common Power Supply and Common Enclosure calculation, SE-C051326701-002 has been updated to reflect this update. The Farley Component Selection Report, PRA-BC-F-11-002, has also been updated to reflect the inclusion of these panels to the UNL list.
CS-B1-02	The Farley breaker Coordination calculations use cable length as part of the justification for proper	E-068 identifies cases where the cable lengths of electrical loads were credited to demonstrate selective coordination for the Cable Spreading room. This assumption is only valid for the Appendix R fire where the	Resolved	This F&O is resolved. An analysis was completed that reviewed the panels that credited cable length as part of the

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
	coordination. This is not a justifiable disposition for use in the Fire PRA.	equipment and cables are assumed damaged for the entire fire area. Supporting Requirement CS-B1-01 Category II requires all buses credited in the Fire PRA to be analyzed for proper over current coordination and protection. Analyze impacted PRA buses for proper coordination and incorporate results into the Fire PRA model.		justification for coordination. The entire function of these panels was then failed for any fire that impacted the cable within the identified length. Once the length requirement was met the function of that cable was the only function failed. A modification is also scheduled to improve coordination for six additional 125VDC load distribution panels per unit. For further information on the modification of these panels see Plant Modifications Committed in Table S-2 of Attachment S of NFPA 805 submittal.
FQ-A3-01	Appendix L of NUREG-CR/6850 had been incorrectly applied to the Main Control Board scenarios in the Farley Fire PRA. The ignition frequencies have since been updated to accurately apply Appendix L.	A non-suppression probability of 3.04E-5 is used for the Main Control Room (NSP-0401* basic events). A review of the Scenario Development report, the Summary Report, and the MCR Report did not locate the justification of this probability. Based on discussion with the Farley team, the values were derived from NUREG/CR-6850, Attachment L. A review of that Attachment did not support a NSP below 1E-4 under the best of circumstances. A NSP of 2E-2 (similar to other NSP events) would make MCR fire the highest contributor to plant risk. Re-evaluate the NSP used for the Control Room and document the evaluation clearly in one of the reports.	Resolved	This F&O is resolved. The Farley MCR analysis has been updated to accurately apply the non-suppression factors as appropriate to the Main Control Board scenarios. The Farley Fire Scenario Report discusses the scenario development process for the Main Control Board and the use of Appendix L in section 13.1.2 of PRA-BC-F-11-014.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
FQ-C1-01	Possible combination events were missing from the cutset results for the Fire PRA model.	A review of cutsets for different sequences found multiple HRA combinations that are not being replaced by a COMBO* event and do not appear to be evaluated for dependence. One such combination is 1OPMSO32-IH-F and OAR_B_1------H-F which has a combined failure probability of approximately 7E-5. A review of the HRA Calculator package supplied shows that no evaluation was performed for this combination of events. Other HRA combinations could also be missing, particularly with new operator actions added for the fire scenarios. HRA dependence could significantly increase cutsets since the rule file makes HRAs independent unless the events are replaced by an evaluated combination. Review the FPRA cutsets without recovery (all events set to screening values) to ensure that all important combinations are evaluated.	Resolved	This F&O is resolved. Every COMBO event is evaluated and incorporated in the fire PRA. An updated dependency analysis was completed after the peer review findings were addressed in the model. The latest results of the dependency analysis captures all important combinations. This is documented in the Human Reliability Analysis for Fire Events, PRA-BC-F-11-016.
FQ-D1-01	The CCFP for Farley Fire PRA was much greater than what the FPIE number was. After continued refinement the Fire PRA CCFP has decreased to a more reasonable value as compared with the FPIE.	In Section 3 of the Farley Nuclear Plant Summary Report, Farley reports a CDF of 9.65E-05/year and a LERF of 1.92E-5/year. This yields a Conditional Containment Failure Probability (CCFP) of 1.99E-01. For the FPIE PRA, the reported CDF was of the order of 3.5E-05/year and the reported LERF was of the order of 2E-07/year. This translates to a CCFP of about 4E-03. This is a significant difference, especially when considering that the leading contributor to LERF for the FPIE PRA, SGTR, is not applicable for fire. This yields inconsistent results.	Resolved	This F&O is resolved. The Farley Fire PRA has continued to evolve and be refined throughout the analysis. Currently the CCFP is at a much more reasonable value based on the final CDF and LERF results. The results and insights related to CDF and LERF can be found in the Farley Summary Report section 3 of PRA-BC-F-11-017.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	Topic	Finding Description	Status	Disposition
		While the current results may be correct, Farley needs to look at the contributors to LERF to explain the basis for the high CCFP with respect to the FPIE PRA CCFP. Farley should look at sequences where the fire not only causes core damage but also directly affects containment integrity. Two likely candidates are sequences that lead to a new ISLOCA scenario and sequences that lead to containment isolation scenarios.		The high fire-induced CCFP was directly related to the human error, OCI_A_1 ------H-F Operator fails to manually isolate containment prior to core damage during a fire event) having a screening HEP of 1.0 in the model reviewed by the peer reviewers. The HEP has been updated by performing a detailed HRA after the peer review and the updated HEP is now 1.20E-02. With new HEP of 1.20E-02, the fire-induced conditional containment failure probability is estimated to be 0.024 from the fire CDF and LERF, 5.24E-5/yr and 1.26E-6/yr, respectively for Unit 1. As presented in Table W-1, Attachment W of NFPA LAR dated September, 25 2012, internal events CDF and LERF for Unit 1 are 1.06E-5/yr and 1.24E-7/yr, respectively. These risks yield an internal events conditional containment failure probability of 0.012, which is comparable to the fire-induced CCFP
FQ-E1-01	The Farley Fire PRA documentation did not accurately address	The summary report lists and describes significant contributors to core damage and LERF. The back references require	Resolved	This F&O is resolved.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
	the types of reviews that were performed during the scenario cutset review sessions.	consideration of analysis issues which are not described in the report as having been done. For example, the back references require a review the results of the PRA for modeling consistency, a review of results to determine that the flag event settings, mutually exclusive event rules, and recovery rules yield logical results, a review of contributors for reasonableness and a review of the importance results for reasonableness. Appendix F notes that these were accomplished and typically refers back to Appendix C. Appendix C does not describe these reviews as being accomplished, nor does it describe the results of the reviews. In addition, back Reference D5 requires a review of non-significant cutsets for reasonableness. Appendix F states that dominant cutsets were reviewed and those that were reduced in frequency to non-significance as a result of the review constitute the review of non-significant cutsets. This does not satisfy the requirement to review non-significant cutsets. Non-significant cutsets generated in the solution of the model need to be reviewed to confirm that their frequency is not underestimated due to modeling errors. Expand the discussion of model solution and review in the summary report to indicate that required review items have been accomplished.		The Farley Summary report includes additional details describing the types of reviews that were completed on the Farley Fire PRA. The type of review and the detailed cutset reviews are described in section C.1 of Appendix C in the Summary Report.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
FQ-F1-01	Level of detail describing the risk significant scenarios was identified as not being sufficient in detail	The documentation of the FPRA results does not adequately describe the top risk contributors such that it is clear why these scenarios, basic events, and human actions are dominant. Based on other findings (FQ-D1-01 and FQ-E1-01), it is not clear that the Farley team understands the bases for these top scenarios. Results presentation is important for PRA acceptability. Understanding of the PRA results is necessary for performing any RI application to support the plant. Provide more detailed discussions of the fire impacts and results to represent a strong understanding of the fire scenarios.	Resolved	This F&O is resolved. The Farley Summary report has been updated to reflect the insights by reviewing the top contributors for CDF and LERF. This describes the fire induced impacts as well as the random failures. The resolution of this finding is found in Appendix C of Farley Fire PRA Summary Report.
FSS-A2-01	Target set definition in Fire Zones (FZs) that do not have fire rated boundaries on all sides as it relates to scenarios that are classified as full room burnouts.	FNP is missing the basis for not including targets outside the fire compartment for full room burnout scenarios. For full room burnout scenarios, all targets in the fire compartment are included. However, there is no documented basis for not including targets outside the fire compartment for full room burnout scenarios. If the compartment has an opening to an adjacent compartment, it was not verified that targets in the adjacent compartment would be outside of the ZOI of all the ignition sources in the compartment analyzed for full room burnout. See F&O PP-B3-01 (F) for a possible resolution.	Resolved	This F&O is resolved. A review of the full room burnout scenarios was completed that looked for open boundaries to the adjoining FZs and the possible interactions that could take place. For some particular fire areas a scenario was postulated that would fail all targets within the fire area. However, in most cases it was determined that there was no ignition source near the open boundary that would impact targets in an adjoining FZ. The Farley Fire Scenario Report includes discussion of the

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
				scenario development process for these specific cases in section 3.1.1 of PRA-BC-11-014.
FSS-B2-01	The Main Control Room Abandonment calculation identifies the potential for a workstation fire but does not describe the fire type in significant detail.	An office workstation fire scenario is discussed in the documentation, but is not fully justified. The workstation fire scenario is potentially the most significant fire scenario considered. Provide better documentation of how the workstation fire was modeled and the results of this fire scenario.	Resolved	This F&O is resolved. The Farley Main Control Room Abandonment Calculation includes the discussion of the workstation fire in section B.8 as a sensitivity to the analysis with the results shown in Table B-8. NUREG/CR-6850 does not provide any basis for this type of fire from an ignition frequency standpoint. Therefore it is not included as one of the potential ignition sources in the base calculation. A review of the sensitivity analysis involving the workstation shows that the analysis is not sensitive to that type of fire given the design of the Main Control Room envelope.
FSS-C1-01	The Farley Fire PRA does not employ the use of a two point fire modeling treatment in the development of the fire scenarios.	Two-point fire intensity model that encompass low likelihood, but potentially risk contributing, fire events were not used in all cases. Fire scenarios were done with ignition sources characterized with one fire intensity. To reach Capability Category II, use a two-point intensity model for all ignition sources. **Utility Comment:** The development of fire scenarios for the Farley Fire PRA did not	Resolved	This F&O is resolved. Although the finding still stands and the SR is met at CC I, further resolution of this F&O will not impact the RICT calculations which are based on a risk delta. The development of fire scenarios for the Farley Fire PRA did not identify any instances where further analysis resolution would

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		identify any instances where further analysis resolution would be gained by the treatment as inferred by the requirements for CC II and CC III. The implications of retaining the CC I treatment in lieu of refining as described for CC II or CC III is potentially a higher calculated CDF contribution. The CC I treatment inherently will not result in under-estimation of fire risk. As such, the current treatment is conservative. Provided this treatment does not result in masking of risk increases in future applications, further refinements are not considered necessary. **Response:** The SR stipulates that a two-point model is required for CC-II. As you stated in your comment, Farley feels that the one-point model is conservative and justified. This would be viewed as the proposed resolution, but the F&O stands.		be gained by the treatment as inferred by the requirements for CC II and CC III. The implications of retaining the CC I treatment in lieu of refining as described for CC II or CC III is potentially a higher calculated CDF contribution. The CC I treatment inherently will not result in under-estimation of fire risk. As such, the current treatment is conservative. Provided this treatment does not result in masking of risk increases in future applications, further refinements are not considered necessary.
FSS-C2-01	The Farley Fire PRA did not characterize the ignition source intensity for a time-dependent growth rate in the scenario development.	Ignition source intensity were characterized such that fire is initiated at full peak intensity and ignition sources that are significant contributors to fire risk were not characterized using a realistic time-dependent fire growth profile. Generic methods from the Hughes Associates Generic Fire Modeling Treatments were used to characterize ignition source intensity. These generic methods did not incorporate fire growth curves. Characterize ignition sources that are significant contributors to fire risk using a realistic time-dependent fire growth profile.	Resolved	This F&O is resolved. Although the finding still stands and the SR is met at CC I, further resolution of this F&O will not impact the RICT calculations which are based on a risk delta. The only readily available reference for a time dependent growth rate that could be considered in the analysis is 12 minutes as recommended in NUREG/CR-6850. The treatment would involve a t^2 growth rate. If a

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		Utility Comment: The only readily available reference for a time dependent growth rate that could be considered in the analysis is 12 minutes as recommended in NUREG/CR-6850. The treatment would involve a t2 growth rate. If a particular source/target interaction has a spacing where the target is at the critical damage spacing threshold, such a treatment may provide some benefit as successful suppression with that time period would prevent target damage. However, if the target is located well within the calculated damage distance, the corresponding time to reaching the damage threshold is very short and effectively precludes any meaningful credit for suppression. In the case of the Farley Fire PRA, the majority of the target spacing for the dominant risk contributors is such that no meaningful credit for suppression is available. In other dominant risk contributors, the scenario involves high energy arcing fault (HEAF) events were no growth time is applicable. The implications of retaining the CC I treatment in lieu of refining as described for CC II/III is potentially a slightly higher calculated CDF contribution. The CC I treatment inherently will not result in under-estimation of fire risk. As such, the current treatment is conservative. Provided this treatment does not result in masking of risk increases in future applications, further refinements are not considered necessary.		particular source/ target interaction has a spacing where the target is at the critical damage spacing threshold, such a treatment may provide some benefit as successful suppression with that time period would prevent target damage. However, if the target is located well within the calculated damage distance, the corresponding time to reaching the damage threshold is very short and effectively precludes any meaningful credit for suppression. In the case of the Farley Fire PRA, the majority of the target spacing for the dominant risk contributors is such that no meaningful credit for suppression is available. In other dominant risk contributors, the scenario involves high energy arcing fault (HEAF) events were no growth time is applicable. The implications of retaining the CC I treatment in lieu of refining as described for CC II/III is potentially a slightly higher calculated CDF contribution. The CC I treatment inherently will not result in under-estimation of fire risk. As such, the current treatment is conservative.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		Response: The Farley modeling was found to be consistent with CC-I but did not meet the requirements of CC-II. The comment provides the basis for stating that the existing treatment is adequate. It does not provide evidence that a time-dependent heat release rate model was used.		Provided this treatment does not result in masking of risk increases in future applications, further refinements are not considered necessary.
FSS-D1-01	The treatment of Secondary combustibles was not clearly defined in the scenario development documentation.	The fire modeling tools selected for use are appropriate for evaluating the zone of influence associated with individual fixed and transient ignition sources, but do not provide for estimating fire growth and damage behavior for fire scenarios involving ignition and fire spread on secondary combustibles. With the generic fire modeling treatment selected for this fire PRA, there does not appear to be a way to model fire growth on secondary combustibles. Consequently, the extent of fire development cannot be modeled. Where secondary combustibles are located within the zone of influence, develop methods for estimating fire growth on secondary combustibles and the damage caused by this additional fire development.	Resolved	This F&O is resolved. The Farley Scenario development notebook was updated to include additional details on how the treatment of secondary combustibles is dealt with during scenario development. Further information regarding this finding can be found in section 4.0 of Farley Fire PRA Scenario Development Notebook
FSS-D7-01	The Fire PRA credits the in cabinet CO2 system installed at Farley. There was no documentation provided to support the availability of this system.	SR FSS-D7 requires credited fire suppression systems to be installed and maintained in accordance with applicable codes and standards, and the credited systems must be in fully operational state during plant operation. These requirements are not met, but fire suppression systems are still being credited. As noted in the Conclusions section	Resolved	This F&O is resolved. Supporting documentation has been included in the Farley Fire Scenario report to further discuss the in cabinet CO2 suppression system and the associated test and inspection procedures that

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		of Document # 0005-0012-002-002-04 (Hughes Associates), "The other main concern with the systems installed at FNP is the periodic maintenance and subsequent corrective action. Firstly, the plant procedures for inspection, testing and maintenance (ITM) do not address a few key activities required by NFPA 12. Secondly, the prioritization of work orders sometimes results in extended impairments (e.g., observed CR / work request tags over two years old), which negatively affects the fire protection program objective to maintain working systems." Credit is being taken for fire suppression systems that do not meet the requirements of FSS-D7 for taking this credit. Verify that credited fire suppression systems are installed and maintained in accordance with applicable codes and standards and demonstrate that credited systems are in a fully operable state during plant operation.		are credited in the Fire PRA. It has also been identified that the system does require modifications, such as mechanical equipment and detection upgrades, to be made to make the system operable as designed. This is found in section 8.1.1 of Farley Fire PRA Scenario Development Notebook
FSS-D7-02	The non-suppression probability that was originally used to calculate the MCR abandonment frequency was unconservative based on direction provided in Appendix P of NUREG-CR/6850.	In Tables 13-1 through 13-12 the equation e^(-lambda*t) was used to calculate the non-suppression probability for MCR abandonment scenarios. The control room lambda value from Table P-2 was selected. The time, t, was obtained through the CFAST runs and plugged into the equation. In scenarios in which the time to abandonment was greater than 25 minutes a nominal NSP of 0 was selected. A NSP of 0 should not be assumed for these cases. Instead, it is suggested to run the CFAST cases longer	Resolved	This F&O is resolved. The application of the MCR abandonment non suppression probability has been re-evaluated using the floor value of 1.00E-03 for all bins that are determined to reach the abandonment threshold. The results of this review are identified in section 13 of Farley Fire PRA Scenario Development Notebook

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		than 25 minutes such that the analysis can credit a larger time with no abandonment conditions reached (i.e. if the case is ran to 60 minutes with no abandonment conditions reached, it can be credited up to 60 minutes) and still use the e(-lambda*t) equation to calculate NSP. Additionally, the MCR equipment rooms are normally unoccupied and NSP should be associated with the electrical equipment room vs. the control room. If the control room lambda is used, a basis should be developed why the control room lambda is more appropriate than the electrical cabinet lambda. If the control room lambda basis has been justified, then a sensitivity analysis should be performed using the lambda of electrical fires. This calculation can be non-conservative.		
FSS-D8-01	The Farley Fire PRA does not look at the time available for a suppression system to successfully suppress a fire before target damage.	Note 8 associated with SR FSS-D8 suggests consideration of the time available to suppress a fire prior to target damage and specific features of physical analysis units and fire scenarios under analysis that might impact suppression system activation and coverage. Such consideration is not documented. Credit is taken for automatic fire suppression in some scenarios without consideration of the factors required under this SR. Perform an analysis that considers the time available to suppress a fire prior to target	Resolved	This F&O is resolved. The Fire PRA was first developed without credit for suppression or detection, the target set for a given scenario was based on the ignition source type. Further in the analysis credit for the existing detection and suppression, and in some cases plant modifications, systems were credited. For these cases where the credit was taken the target set was not changed based on the time to suppression

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		damage and the specific features of the PAUs and fire scenarios under analysis to determine what impact they have on suppression system activation and coverage.		or distance to target. Instead a conservative approach was taken to leave the original target set included in the Fire PRA along with the failure rate of the suppression system, therefore not requiring a review of damage time vs. suppression time. This is found in section 8.0 of Farley Fire PRA Scenario Development Notebook.
FSS-E3-01	The Farley documentation did not address the uncertainty related to the use of fire modeling for the fire scenarios.	Supporting requirement E3 asks to provide a mean value of, and statistical representation of, the uncertainty intervals for the parameters used for fire modeling the fire scenarios. Farley performed fire size and heat release rate selection in accordance with NUREG/CR-6850 and/or applicable FAQs. However, the methods for developing the statistical representation of the uncertainty intervals and mean values currently do not exist. However, this is not reported in the documentation. In the documentation, explain that it is understood that methods for developing the statistical representation of the uncertainty intervals and mean values currently do not exist. **Utility Comment:** This specific F&O was issued against a technical element and the indicated resolution involves a documentation	Resolved	This F&O is resolved. Although the finding still stands and the SR is met at CC I, further resolution of this F&O will not impact the RICT calculations as the F&O pertains to a documentation issue. The documentation has been updated to include discussions related to the uncertainty for fire modeling. See Table D-1 of the Farley Fire PRA Summary report. The associated SR was dispositioned as CC I which is judged to be sufficient given the two concerns noted.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		clarification. This documentation clarification will be implemented.		
FSS-F1-01	The exposed structural steel evaluation was not originally performed as part of the Farley Fire PRA.	Section 2.11 of the FNP Summary Report (FNP_Summary_Report_final.pdf) claims that, "The Structural Steel Evaluation performed to evaluate the potential for fire to impact structural steel capacity which could impact fire compartment boundaries is documented in the FNP Fire PRA Report PRA-BC-F-11-014, Rev. 0, Fire Scenarios Report." This documentation was not found in the referenced report. Include in the Fire Scenarios report the structural steel evaluation identified in final Summary Report and update self-assessment.	Resolved	This F&O is resolved. A review and analysis was completed of the structures at Farley for both units to determine the amount of exposed structure steel that is susceptible to fire damage and ultimately leading to a building collapse. The analysis concluded that there is a potential for this scenario to occur in the Turbine Building. This scenario has been added and is accounted for in the total plant risk and delta risk calculations. This is found in section 10.5 of Farley Fire PRA Scenario Development Notebook.
FSS-G6-01	The Farley Fire PRA MCA analysis was incomplete at the time of review with many open items.	The Multi-compartment analysis identifies several areas where further evaluation is required. This evaluation has not been completed to either screen the zone or develop a fire scenario based on multi-compartment fire. A screening of the multi-compartment scenarios were done, those that were screened out were not included in the quantification. The multi-compartment scenarios flagged for further evaluation are in Table 3-1 of the Multi-Compartment Analysis. Further evaluation is still being worked on, so these scenarios have not been included in quantification. Given the current CDF, the MCA could increase risk above 1E-4/yr.	Resolved	This F&O is resolved. The Farley Fire PRA MCA analysis has been completed with all scenarios being evaluated. The HGL/MCA report has been updated to show the final results for the analysis. This is found in Attachment B of PRA-BC-11-015.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		Complete the MCA to either quantify the PAUs where a fire could spread to an adjacent PAUs or screen the PAUs for MCA		
FSS-H1-01	Non-Fire PRA targets were removed from the database leading to inconsistencies between the scenario development sheets and what was identified in the field.	For fire scenarios considered during the peer review walkdown, the nature and characteristics of the damage target set were different in three different sets provided for review, including the computer printout of the fire scenario summary and two sets of walkdown notes. One consistent set of documentation should be maintained in a retrievable format. Include all relevant target sets in the computer-based documentation and handle by disposition those targets that are not risk significant for a particular scenario.	Resolved	This F&O is resolved. The scenario development database has been re-populated with all target set information, targets specifically modeled in the Fire PRA and those that are not. The scenario printout sheets found in Appendix A of the Fire scenario development report contain all targets identified during the walk down phase regardless of the relationship to Fire PRA components. This is found in Appendix A of Farley Fire PRA Scenario Development Notebook.
FSS-H5-01	The Farley documentation did not address the uncertainty related to the use of fire modeling for the fire scenarios.	The generic fire modeling tool referenced in the Fire Scenario Report, Reference 6 (Hughes generic treatment) is used for generic treatment of ignition sources as an approach to bound many scenarios, but its use does not provide uncertainty treatment on a fire scenario basis. Provide uncertainty evaluations at least generically for those scenarios that use the generic treatment tools and on a case by case basis for the sources that use additional detailed fire modeling to further describe the scenarios used.	Resolved	This F&O is resolved. Although the finding still stands and the SR is met at CC I, further resolution of this F&O will not impact the RICT calculations as the F&O pertains to a documentation issue. The documentation has been updated to include discussions related to the uncertainty for fire modeling in response to F&O FSS-E3-01. See Table D-1 of the Farley Fire PRA Summary report.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		Utility Comment: This specific F&O is inconsistent with F&O FSS-E3-01. The indicated resolution for FSS-E3-01 states in part that the analysis documentation should be enhanced to note that methods for developing the statistical representation of the uncertainty intervals and mean values currently do not exist. However, F&O FSS-H5-01 then asks to undertake evaluations to address uncertainty. This latter F&O should be revised so that it is consistent with FSS-E3-01. **Response:** The F&Os address the specific SR requirements. The response to F&O FSS-E3-01 may be used to justify the treatment of uncertainty for FSS but the F&O documents compliance with the standard and as such remains.		
IGN-A7-02	Newly installed potential Ignition sources were identified in the field that were not included as part of the original scenario development.	During the walkdown - ignition sources (specifically electrical cabinets) were found in the plant that is not listed on the list of ignition sources for the particular PAU. Specific examples include N1R1L0001 in the cable spreading room and N1R15A002X and N1R5A003X in the switchgear room. A walkdown and/or review of plant modification is necessary to ensure the plant FPRA reflects the as built as operated configuration. This issue may be due to new plant equipment that was added after the initial ignition frequency walkdown – nevertheless	Resolved	This F&O is resolved. The Farley Fire PRA has been in development for some time. The ignition source walk down and scenario development were some of the first tasks that were completed as part of this analysis. A qualitative review of the panels identified during the peer review walkdown showed no significant change in the plant CDF. This is based on the fire zones these panels were located in and the

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		the fire PRA should be reconciled to include these new ignition sources.		level of scenario development already included in these fire zones. The panels are located in a part of the room that already contains detailed scenarios and the introduction of the new sources are not expected to change the target set of adjoin scenarios. Section 3.5 of the Summary Report, provides steps that will be taken to account for changes in the plant design that have occurred since the initial Fire PRA development.
IGN-A7-04	The yard transformers had been incorrectly binned during the Task 6 development and should be moved to their appropriate bins.	Bins 27-29 have not been filled. Large Yard Transformers have been incorrectly binned in Bin 23 (“indoor transformers”). It is clearly stated in NUREG/CR-6850 that large yard transformers are not part of this count. As a result each large outdoor transformers (MT, UAT, SuT) should be binned in both Bin 27 (Yard Transformer – Catastrophic) and Bin 28 (Yard Transformer – Non Catastrophic). Additionally, Bin 29, Transformer Yard – Others, should also be filled. Since Bin 23 may have been misinterpreted, it is suggested that indoor transformers typically associated with essential lighting, etc. be looked at for applicability in the FPRA if not already evaluated. Indoor transformers	Resolved	This F&O is resolved. The Farley Fire PRA task 6 has been updated to accurately represent the transformers located in the yard to their applicable bins and have been removed from bin 23. The frequency per component has been updated accordingly and used for the applicable scenarios. See Appendix C of Plant Partitioning and Fire Ignition Frequency for Farley Fire PRA, PRA-BC-F-11-009.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		over 45kVA should be included in the count for this bin.		
IGN-A7-05	The Farley scenario development did not accurately account for the frequency split between the two fire zones as it was identified in the field.	During the walkdown of the Bravo 4160 Switchgear room, it was observed that the Foxtrot Switchgear was split between 2 PAUs. The switchgear had a count of 15 vertical sections. PAU 335 had a count of 8 switchgear vertical sections and PAU 343 had a count of 8 vertical sections. This is a clear example of inadequate PAU boundary. Recommend that the PAU such that the Foxtrot switchgear is contained in one PAU and the count of the entire switchgear should be 15 vertical sections. In cases where ignition sources have been split between PAUs the count should be verified correct.	Resolved	This F&O is resolved. The Farley Fire PRA has been updated to accurately correct the scenario development to account for the ignition source split between the two fire zones of the SWGR room. The ignition source count of the SWGRs has not been changed to reflect the accurate number of cubicles. This change would result in a non-significant impact to the total plant ignition frequency based on the total count for Bin 15. The ignition frequency for the scenarios related to the SWGRs are accurately represented. These updates can be found in the Farley Scenario report, Appendix A, PRA-BC-F-11- 014.
IGN-A9-01	The transient factors in the ignition frequency development had identified fire zones that had a 0 factor which led to a frequency of 0.	PAU 2321 (Sample Panel Room) has a transient fire frequency of zero. Similar to the first page of Appendix B, a storage factor of “low” or 1 should be chosen such that 2321 has a non-zero transient fire frequency. Right now 2321 has a non-zero ignition frequency due to a small number of cable in the area filling Bins 11 and 12. A non-zero transient factor should be filled in.	Resolved	This F&O is resolved. The transient ignition frequency allocation has been re-visited for the Farley Fire PRA based on this finding. The appropriate changes have been made to accurately reflect the transient ignitions sources located within each fire zone. These updates were made

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
				in Farley Plant Partitioning and Ignition Source Task 1 and 6 report, PRA-BC-F-11-003, and carried into the ignition source calculation for the scenario development, PRA-BC-F-11-014.
PP-B3-01	The Farley Fire PRA did not contain sufficient information on scenario development with respect to the crediting of fire barriers.	SNOC has not provided sufficient evidence that Fire Zone PAUs were evaluated for fire resistance capabilities of barriers, nor was there sufficient evidence that credited spatial separations were analyzed. Specific examples are cited in PRA-BC-F-11-001, Section 2.2, for PAUs that use "natural divisions." The document cites that the lack of fire barriers between these PAUs will be evaluated during the MCA. However, the MCA analysis appears to only discuss hot layer issues, and does not consider whether a fire propagates outside of the PAU or if there is a zone of influence and target damage outside of the PAU. Another example of where spatial separation is credited is Tool Room 0441. Full room burnout scenarios are developed and quantified, but without sufficient evidence that fire barriers or spatial separation issues have been evaluated. It appears that specific PAUs are screened from having multi-compartment impacts without consideration of fire propagation or ZOI impact across spatial divisions.	Resolved	This F&O is resolved. The Farley scenario development report has been updated to provide more details on the scenario development based on the ignition source and target identification process. This can be found in the Farley Scenario Development report, PRA-BC-F-11-014, section 3.1.1. The impact on the Hot Gas Layer and Multi-Compartment Analysis has also been revisited to assure that the boundaries of the rooms have been adequately represented in the calculation of the volumes.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		SNOC has presented a plan to resolve the Fire Zone PAU vs. Fire Area PAU issue. Implementation of this plan is sufficient to address the issues identified in PPB2 and PP-B3. In the plan, Fire Areas will be treated as PAUs. Particularly, SNOC staff have acknowledged that for "full burn" and "base case" fire scenarios, they will review and document the capabilities of barriers and the appropriateness of credited spatial separations, and will not inappropriately credit barriers or spatial separations for fire scenarios. The plan includes the following: 1. Those APs that have one or more boundaries that are not physical features or are not rated fire barriers will be identified and a requirement will be added to clarify that this must be recognized in the development of fire scenarios. There will be confirmation that the results of the above have been observed and documented. 2. Enhance the documentation to acknowledge the crediting of non-rated physical boundaries and provide a basis recognizing that the justification will rely on physical observations during plant walkdowns or through equivalent means as well as general construction methods (masonry block wall, concrete walls, etc.). 3. Address the nature and consequence of anticipated fire events for all APs for which		

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		explicit fire scenarios are not developed (base cases) and confirm that the results are appropriate given the boundaries for the AP. 4. Confirm that bounding room burn-out cases are not used for any APs that are not fully bounded by physical fire barriers, and that there is a justification for crediting those physical barriers. 5. Confirm that the resulting analysis does not change (reduce) the level of resolution associated with the existing fire scenarios developed to support the requirements of SRs associated with FSS. Modify the hot gas layer and multi-compartment analysis (MCA) so that any unnecessary conservatism caused by using a smaller volume artificially caused by an assumed AP boundary are removed.		
PP-C3-01	The Farley Fire PRA did not contain sufficient information on scenario development with respect to the identification of fire barriers.	Plant personnel have given verbal assurance that plant walkdowns have been performed to confirm the plant partitioning boundaries. It is reasonable to presume that the fire protection engineer would perform this walkdown task. In addition, walkdowns were performed to support the Fire PRA ignition frequency task. Furthermore, some notes were found as further evidence that some walkdowns were performed. However, documentation of the plant partitioning walkdown is not readily available for peer review. SR PP-C3 requires documentation of key or unique features of	Resolved	This F&O is resolved. The Farley Task 1 and 6 report identifies the ignition sources identified in each fire zone. The results of the walkdowns are input into a database that contains the necessary information related to Task 1 and 6. This database is considered to be the controlled copy of the results of these tasks. These results are found in Appendix D of report PRA-BC-F-11-009. Section 3.1.1 of the

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		the partitioning elements for each physical analysis unit. SR PP-B7 requires a confirmatory walkdown of partitioning elements. Include Plant Partitioning walkdown sheets as part of PRA secondary documentation, and refer to the walkdown sheets in PRA-BC-F-11-001, Farley Fire PRA Tasks 1 & 6, Plant Partitioning and Fire Ignition Frequency. In particular, fire barriers and spatial separations that are credited in fire scenarios should be validated. When where no prior documentation can be found, new walkdowns may be required.		Farley Scenario Report, PRA-BC-F-11-014, describes the process of identifying applicable scenarios based on the ignition source, surrounding targets and fire barriers.
PP-C3-02	The Farley Fire PRA did not contain sufficient information on scenario development with respect to the documentation of fire barriers.	Fire Zones are identified as Fire PRA plant analysis units in PRA-BC-F-11-00. Fire PRA staff have expressed that the Fire Areas, not Fire Zones, should be assessed as the PAUs. However, the Fire Zone PAU form the basis for initial PAU ignition frequency, whole room burns, and initial screening in later PRA analysis Fire Zones as PAUs are used consistently and extensively in the FPRA documentation. There is a disconnect between the PAUs defined in PRA-BC-F-11-00 and SNOC staff's statements of what constitutes a PAU. This adversely affected the review of the Plant Partitioning technical element. SNOC desires to call the entities that are currently described as Fire Zone PAUs as Administrative Partition, and to treat Fire	Resolved	This F&O is resolved. The Farley Fire PRA documentation has been updated to be consistent in the naming convention throughout the analysis concerning the use of PAU and fire zone. The ‘rooms’ at Farley are considered fire zones, while the fire areas are considered PAUs. The Task 1 and 6report, Plant Partitioning and Ignition Frequency PRA-BC-F-11-009, Cable selection and Detailed Circuit Analysis PRABC-F-11-003, and the Farley Scenario report contain this clarification.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		Areas as PAUs. F&O PP-B3-01 identifies an acceptable plan to address the technical issues around the definition of PAUs, that Fire Areas, not Fire Zones, form the basis for PAUs. Fire Zones and similar entities will be identified as "Administrative Partitions" (AP). Since the term "Physical Analysis Unit" or PAU is extensively in Fire PRA documentation to describe Fire Zone PAUs, all Fire PRA documents should be reviewed and revised to call these compartments Administrative Partition. Furthermore, the term "Administrative Partition" (AP) should be defined in the PP documentation and the APs descriptions (formally, Fire Zone PAUs), should be retained.		
PRM-B2-01	The Farley internal events finding had only been partially addressed in respect to the impact on the Fire PRA.	Internal Events PRA peer review exceptions and deficiencies have only partially been dispositioned. Table 1 of the Fire Model document (PRA-BC-F-11-004_V0a) lists some of the internal events findings, but not all. All findings included in the internal events peer review must be included and disposed in the PRM notebook. Disposition of findings could not be verified. Discussion with Southern Company personnel indicated that some of the findings had not been addressed. Expand Table 1 of the Fire Model document to include all findings. Describe the impact of the finding on the fire PRA. For those that	Resolved	This F&O is resolved. Table 1 of Fire PRA logic Development, PRA-BC-F-11-004 has been updated to address all internal events PRA findings and their impacts on the fire PRA.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		impact model elements applicable to the fire analysis, describe the resolution in sufficient detail to allow a reviewer to conclude the finding has been dispositioned.		
PRM-C1-01	The RCP shutdown seals were not adequately discussed in the documentation for the Fire PRA model development.	The new RCP shutdown seals are included in the fault tree model but are not described in Appendix B. Appendix B should be revised to describe these new seals and their impact on RCP seal failure flow rate. The Fire PRA modeling pertaining to RCP seal failure is not adequately described in PRA-BC-F-11-004. Revise Appendix B to describe the new shutdown seals and their impact.	Resolved	This F&O is resolved. Fire PRA has been developed based on internal events PRA model having model of RCP shutdown seal. Section 2.0, Appendix B of Fire PRA logic Development, PRA-BC-F-11-004 has been updated to add RCP shutdown seal modeling.
UNC-A1-01	The Farley fire PRA provided Train A and B CDF results but did not define total plant CDF. The parametric uncertainty analysis should be more specific in scope and use a greater sampling size.	Farley presents the CDF results in Section 3.0 of the Summary Report. The way the results are presented are as an annualize CDF for Train A operating and an annualize CDF for Train B operating and both are called total plant CDF. There is no discussion as to what these two CDF values meant or a value for the "true" plant CDF. In Appendix D of the Summary Report, Farley presents the results of their parametric uncertainty analysis for CDF. Although not documented, this appears to be for CDF related to Train A Operating only. The parametric uncertainty analysis was performed using the Latin Hypercube method with only 1000 samples. The resulting curve was not well behaved and the calculated mean is well below the point estimate in Section 3.	Resolved	This F&O is resolved. Appendix D of the Farley Summary report has been updated with a revised parametric uncertainty analysis for both CDF and LERF for Train A and B individually. The quality of the analysis was improved by applying the Monte Carlo method with 50,000 samples. The resulting curves are well behaved and the calculated means show minimal difference when compared to the point estimates. Discussion of how the total plant CDF/ LERF is calculated is also provided in the Summary Report. This describes how the Train A and Train B results are averaged

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		As a start, Farley needs to define what the two results, annualize CDF for Train A operating and an annualize CDF for Train B operating, mean and a single total Plant CDF needs to be presented. This will probably be the average of the original two values. For the uncertainty analysis, Farley needs to document what is covered by the analysis, Train A results, Train B results or both. Farley did run an uncertainty case using 10,000 samples and the results seemed to be better behaved. Farley is running an uncertainty case with 50,000 samples which is consistent with their FPIE PRA process. The results of this analysis should be presented in Appendix D in the Summary Report instead of the current analysis.		together to obtain the total plant CDF/LERF.
UNC-A1-02	The Farley documentation did not adequately address the review of LERF scenarios in the analysis to show that the appropriate reviews had been completed.	Farley did quantify the fire-related LERF for Unit 1 but failed to meet the requirements from LE-F2 and LE-F3 from Section 2 which require that "REVIEW contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant-specificity is appropriate for significant contributors, etc.)" and "IDENTIFY and characterize the LERF sources of model uncertainty and related assumptions in a manner consistent with the applicable requirements of Tables 2-2.7-2(d) and 2-2.7-2(e)." As discussed in F&O FQ-D1-01, the calculated LERF and CCFP indicate that there some potential issues with the LERF calculation.	Resolved	This F&O is resolved. The Farley Summary report has been updated to reflect the insights by reviewing the top contributors for LERF. This describes the fire induced impacts as well as the random failures. The resolution of this finding is found in Appendix C of Farley Fire PRA Summary Report.

Table E2-4. Resolution of the FNP Fire PRA Peer Review Findings				
F&O #	**Topic**	**Finding Description**	**Status**	**Disposition**
		See F&O FQ-D1-01 (F) and perform the reasonableness reviews after requantifying.		

5.0 General Conclusions Regarding PRA Capability

The information provided in this enclosure demonstrates that the FNP at-power internal events PRA model (including flooding) and the fire PRA model conform to the standard at CC-II which satisfies the guidance of RG 1.200, Revision 2. In addition, the FNP PRA model complies with all requirements for technical adequacy of the baseline PRA as defined in NEI 06-09 (Reference 1) as clarified by the NRC final safety evaluation of this report (Reference 2).

The FNP internal events PRA model (including flooding) and the FNP fire PRA model technical capability evaluations described above provide a robust basis for concluding that the PRA models are suitable for use in supporting the License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines".

6.0 References

1. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines", Revision 0-A, October 2012, (ADAMS Accession No. ML12286A322.
2. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,'" dated May 17, 2007 (ADAMS Accession No. ML071200238
3. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 2, March 2009.
4. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 1, January 2007.
5. Regulatory Guide 1.200, "An Approach for Determining Technical Adequacy of PRA Results for Risk-Informed Activities," For Trial Use, USNRC, February 2004.
6. 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.
7. ASME RA-Sc-2007, "ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications", Addenda to ASME RA-S-2002, ASME, New York, NY, August 31, 2007.
8. ASME RA-SB-2005, "Addenda to ASME RA-S-2002 Standard for PRA for Nuclear power Plant Applications, "American Society of Mechanical Engineers, New York, NY, December 2005.
9. ASME RA-Sa-2003, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-S-2002, April 2002 and Addenda to Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers 2003.
10. NEI-00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," 2000.

11. LTR-RAM-II-10-015, “RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Farley Nuclear Power Plant Probabilistic Risk Assessment,” May 2010.
12. LTR-RAM-II-12-007, “Fire PRA Peer Review of the Farley Nuclear Plant Fire Probabilistic Risk Assessment Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard,” March 2012.
13. NEI 05-04, “Process for Performing PRA Peer Reviews Using the ASME PRA Standard (Internal Events),” Revision 2, September 2008.
14. NUREG/CR-6928 / INL/EXT-06-11119, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, February 2007.

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, “Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines”

Enclosure 3

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

Table of Contents

Introduction

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

Attachment 1 to this enclosure presents the plant-specific bounding analysis of seismic risk to FNP. Attachment 2 to this enclosure presents the justification for excluding analyses of other external hazards from the FNP PRA.

Scope

Topical Report NEI 06-09, Revision 0-A (Reference 1) 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, non-mandatory Appendix 6-A in the ASME/ANS PRA Standard (Reference 4) provides a guide for identification of most of the possible external events for a plant site. This information was reviewed for the Farley site and augmented with a review of information on the site region and plant design to identify the set of external events to be considered. The data in the UFSAR (Reference 7) regarding the geologic, seismologic, hydrologic, and meteorological characteristics of the site region as well as present and projected industrial activities (e.g., increases in the number of flights, construction of new industrial facilities) in the vicinity of the plant were also reviewed for this purpose. No new site-specific and plant-unique external hazards were identified through this review and associated plant visit.

Table E3.1
Minimum Scope of External Hazards to be considered

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

The scope of this enclosure is consideration of the above hazards for FNP. Seismic events in particular are considered in Attachment 1, and the other listed external hazards are considered in Attachment 2.

Technical Approach

The guidance contained in NEI 06-09 states that all hazards that contribute significantly to incremental risk of a configuration must be quantitatively addressed in the implementation of the RICT Program. The following approach focuses on the risk implications of specific external hazards in the determination of the risk management action time (RMAT) and RICT for the Technical Specification (TS) Limiting Conditions for Operation (LCOs) selected to be part of the RICT Program. The process includes the ability to address external hazards by 1) Screening the hazard based on a low frequency of occurrence, 2) Bounding the potential impact and including it in the decision-making or 3) Developing a PRA model to be used in the RMAT/RICT calculation.

The overall process for addressing external hazards is shown in Figure E3.1, below, where each hazard identified in Table E3.1, above, is addressed individually.

The process considers two aspects of the external hazard contribution to risk. The first is the contribution from the occurrence of beyond design basis conditions, e.g., winds greater than design, seismic events greater than design-basis earthquake (DBE), etc. These beyond design basis conditions challenge the capability of the SSCs to maintain functionality and support safe shutdown of the plant. The second aspect addressed are the challenges caused by external conditions that are within the design basis, but still require some plant response to assure safe shutdown, e.g., high winds or seismic events causing loss of offsite power, etc. While the plant design basis assures that the safety related equipment necessary to respond to these challenges are protected, the occurrence of these conditions nevertheless cause a demand on these systems that in and of itself presents a risk.

Step 1 – Hazard Screening

The first step in the evaluation of an external hazard is screening based on an estimation of a bounding core damage frequency (CDF) for beyond design basis hazard conditions. As noted in Regulatory Guide 1.200 (Reference 8), the fundamental criteria that have been recognized for screening-out events are the following: an event can be screened out if either (1) it meets the criteria in the NRC's 1975 Standard Review Plan (SRP) or a later revision (Reference 5); or (2) if it can be shown using a demonstrably conservative analysis that the mean value of the design-basis hazard used in the plant design is less than 10^{-5} per year and conditional core damage probability is less than 10^{-1}, given the occurrence of the design-basis-hazard event; or (3) if it can be shown using a demonstrably conservative analysis that the CDF is less than 1E-06 per year. The bounding CDF estimate is often characterized by the likelihood of the site being exposed to conditions that are beyond the design basis limits and an estimate of the bounding conditional core damage probability (CCDP) for those conditions. Sometimes, the bounding CCDP is conservatively assumed to be 1.0. For FNP, however, bounding CDF values are estimated in Attachments 1 and 2, without the estimation of CCDP.

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

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

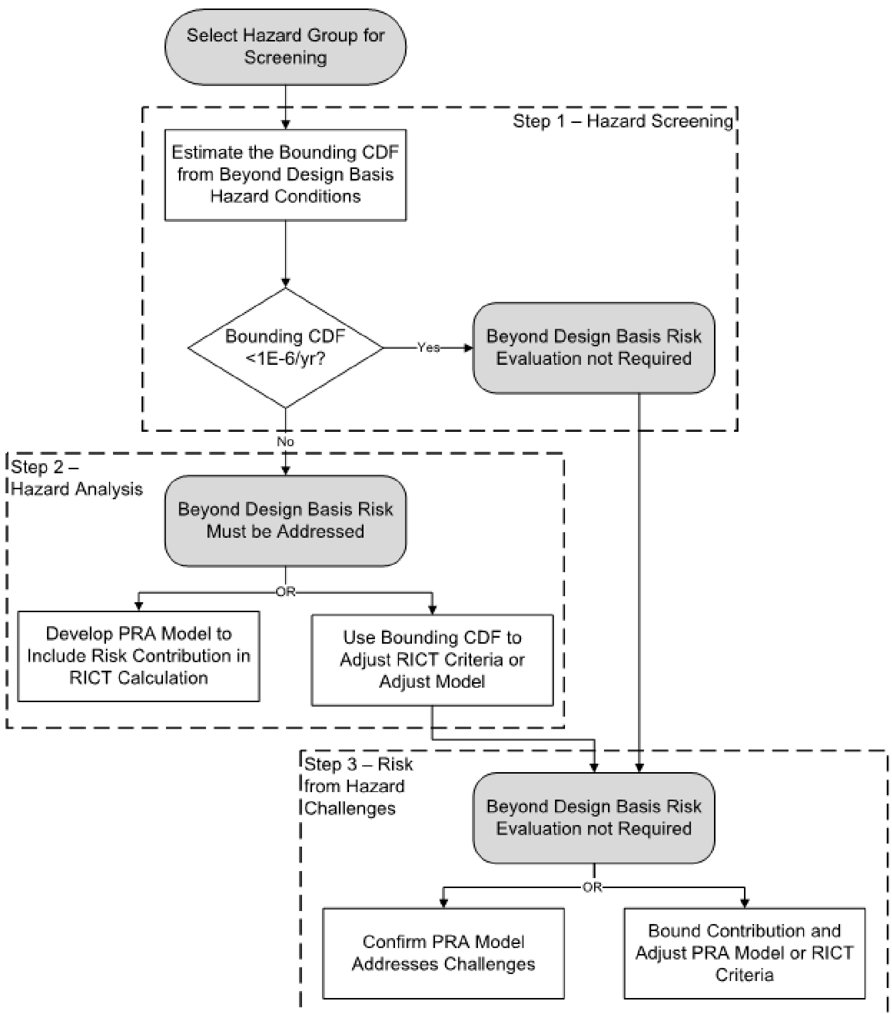

Figure E3.1
Process for Addressing External Hazards in RICT Program

While the direct CDF contribution from beyond design basis hazard conditions can be shown to be non-significant using this approach, some external hazards can cause a plant challenge, even for hazard severities that are less than the design basis limit. These considerations are addressed in Step 3 of the process.

There is one other important consideration for screened hazards that must be addressed within the RICT Program. This consideration relates to maintaining the boundary conditions of the base risk analysis. The screening process described above assumes that the capability of the plant to withstand the hazard is consistent with the design assumptions. In some cases, plant activities can change this assumption. Some examples are shown below:

- Removal of a toxic gas monitor from service on the control room heating, ventilation and air conditioning (HVAC) system can impact the ability of the plant systems and operators to respond to a toxic gas release. For FNP, the Control Room Emergency Filtration/Pressurization System is excluded from the RICT Program; therefore, this boundary condition is not applicable.

- Removal of a tornado missile or flood barrier from service in order to support a maintenance activity can degrade the capability of the plant to respond to such hazards, if the removal of the barrier reduces the protection of equipment that is expected to be available. That is, if the barrier only protects equipment that is considered out of service under the RICT Program, there is no need to address this further, but if other equipment that is intended to be available could be impacted, the basis for the screening of the hazard becomes invalid. For FNP, as a precondition to entering a RICT, plant procedures assure that if the design basis assumptions applicable to a hazard are temporarily not applicable (for example, barrier degradation), which may increase the likelihood of a plant challenge from loss of equipment that is not considered out of service within the RICT Program, appropriate compensatory measures are implemented to accomplish the following:

    - Compensate for loss of protection; or

    - An incremental CDF/Large Early Release Frequency (LERF) equal to the applicable hazard frequency for all impacted equipment will be added to the incremental CDF/LERF resulting from the unavailability of structures, systems, and components (SSCs) attributed to the LCO Condition for which a RICT is calculated.

<u>Step 2 - Hazard Analysis</u>

There are two options in cases where the bounding CDF for the external hazard cannot be shown to be less than 1E-6/yr. Such hazards are generally those with relatively larger frequencies of beyond design basis conditions, such as seismic events. The first option is to develop a PRA model that explicitly models the challenges created by the hazard and the role of the SSCs included in the RICT Program in mitigating those challenges. The second option for addressing an external hazard is to compute a bounding CDF contribution for the hazard. The basic approach to computing a bounding CDF is as follows:

Estimate Bounding CDF

This approach is described in Attachment 1 of this Enclosure for the seismic hazard.

Evaluate Potential Risk Increases Due to Out of Service Equipment

Given the selection of an estimated bounding CDF/LERF, the approach considered must assure that the RICT Program calculations reflect the change in CDF/LERF caused by the out of service equipment. For FNP, as discussed in Attachment 1, the only beyond design basis hazard that could not be screened out is the seismic hazard, and as demonstrated in Attachment 1, with the approach used the change in risk with equipment out of service cannot be higher than the bounding seismic CDF (SCDF).

Evaluate Bounding LERF Contribution

The RICT Program requires addressing both core damage and large early release risk. When a comprehensive PRA does not exist, the LERF considerations can be estimated based on the relevant parts of the internal events LERF analysis. This can be done by considering the nature of the challenges induced by the hazard and relating those to the challenges considered in the internal events PRA. This can be done in a realistic manner or a conservative manner. The goal is to provide a representative or bounding conditional large early release probability (CLERP) that aligns with the bounding CDF evaluation. The incremental large early release frequency (ILERF) is then computed as described in Attachment 1 of this Enclosure.

Step 3 - Risks from Hazard Challenges

Steps 1 and 2 address the direct risks from damage to the facility from external hazards. While the direct CDF contribution from beyond design basis hazard conditions can be shown to be non-significant using Steps 1 and 2 without a full PRA, there are risks that may be unaccounted for. These risks are related to the fact that some external hazards can cause a plant challenge, even for hazard severities that are less than the design basis limit. For example, high winds, tornadoes, and seismic events can cause extended loss of offsite power conditions below design basis levels. Additionally, depending on the site, external floods can challenge the availability of normal plant heat removal mechanisms.

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

Attachment 1 to this enclosure provides an analysis using Steps 1 and 2 for the FNP site with respect to the beyond design basis seismic hazard. Attachment 2 to this enclosure provides an analysis of the representative external hazards for the FNP site, as discussed in Step 3.

References

1. Nuclear Energy Institute (NEI) Topical Report NEI 06-09, “Risk-Informed Technical Specifications Initiative 4b, Risk- Managed Technical Specifications (RMTS) Guidelines,” Revision 0-A, October 2012 (ADAMS Accession No. ML 12286A322.

2. 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.

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. 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.

5. NUREG-75/087, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition,” 1975.

6. NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” June 1991.

7. Joseph M. Farley Nuclear Plant Units 1 and 2 UFSAR, Rev 28 December 2017

8. Regulatory Guide 1.200, “An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities”, Revision 2, U.S. Nuclear Regulatory Commission, March 2009

Attachment 1: Seismic Bounding Analysis

The purpose of this attachment is to present the analysis that bounds the potential seismic impact and include it in the decision-making process (Step 2 from Figure E3.1), as a seismic PRA is not available for FNP. The process for analyzing an unscreened external hazard without the use of a full PRA involves the following three steps:

1. Estimate Bounding CDF

2. Evaluate Potential Risk Increases Due to Out of Service Equipment

3. Evaluate Bounding LERF Contribution

Estimate Bounding CDF

A seismic margin assessment (SMA) was developed for the FNP Individual Plant Examination for External Events (IPEEE) (Reference A.1-1). Thus, there is not a current estimate of seismic core damage frequency (SCDF), so an alternative approach is taken to develop an SCDF estimate. This approach uses the current FNP seismic hazard curve and a limiting seismic capacity of a component whose seismic failure would lead directly to core damage as identified in the SMA. In this approach, a plant level high confidence of low probability of failure (HCLPF) seismic capacity corresponding to the limiting HCLPF component is used to determine the failure probabilities as a function of seismic hazard level, and these are convolved with the seismic hazard curve. This is a commonly used approach to estimate SCDF when a seismic PRA is not available. This approach is consistent with approaches that have been used in other regulatory applications (e.g., Reference A.1-4).

The seismic hazard for the FNP site was evaluated in 2013 (Reference A.1-2) and provided to NRC (Reference A.1-3). The FNP IPEEE (Reference A.1-1) was a limited scope seismic margins assessment performed relative to a review level earthquake (RLE) of 0.1g PGA (peak ground acceleration), and established that the corresponding HCLPF for equipment required for response to the RLE is a HCLPF of 0.1g referenced to peak ground acceleration (PGA), which corresponds to a spectral frequency of 100 Hz. The HCLPF can also be scaled to other spectral frequencies, based on the reference level earthquake used in the SMA. The PGA is generally used as the convolution acceleration for most seismic PRAs, including other SPRAs performed by SNC. However, because this seismic risk estimation is based on the overall plant-level HCLPF, the controlling seismic failures would be unknown. Based on judgment from the recent seismic walkdowns, the seismic risk for FNP could be sensitive to the natural frequency of the service water pond dike/dam, or to electrical cabinets and equipment. The natural frequencies of these items range from about 2.2 Hz to 8 Hz. Therefore, the FNP plant level fragility is estimated for the PGA (100Hz) and for the 2.5, 5 and 10 Hz spectral hazard curves. That is, four convolutions hazard frequency and HCLPF are performed, one for each spectral frequency. The average of these four results is then used to estimate the seismic risk. Note that the HCLPF is judged to be very conservative for FNP based on the following:

- Detailed seismic walkdowns of the FNP structures and equipment have recently been performed for a future seismic PRA. The results of these walkdowns show that virtually all of the equipment reviewed has much higher seismic capacity than 0.1g.
- Other Westinghouse PWRs with equipment similar to FNP have HCLPFs closer to 0.3g.

Therefore, it is judged that there is significant conservatism in using the IPEEE HCLPF of 0.1g for the estimation of the FNP plant seismic fragility.

Calculation of the SCDF in this manner also requires definition of uncertainty parameters for seismic capacity. The uncertainty parameter for seismic capacity is represented by a combined beta factor (βc) of 0.4. This is a commonly-accepted approximation, and is consistent with the value used in other regulatory applications (e.g., Reference A.1-4). Using the above inputs, the total estimated FNP SCDF is determined to be 4.51E-6 (Reference A.1-7).

Therefore, a RICT bounding value of 4.51E-6 will be used as the estimate of SCDF ($ICDF_{seismic}$) for the LAR submittal RICT calculations.

Evaluate Potential Risk Increases Due to Out of Service Equipment

The approach taken in the computation of SCDF in Reference A.1-5 assumes that the SCDF can be based on the likelihood that a single seismic-induced failure leads to core damage. This approach is bounding and implicitly relies on the assumption that seismic-induced failures of equipment show a high degree of correlation (i.e., if one SSC fails, all similar SSCs will also fail). This assumption is conservative, but direct use of this assumption in evaluating the risk increase from out of service equipment could lead to an underestimation of the change in risk. However, if one were to assume no correlation at all in the seismic failures, then the seismic risk would be lower than the risk predicted by a fully correlated model, but the change in risk using the un-correlated model with a redundant piece of important equipment out of service would be equivalent to the level predicted by the correlated model.

If the industry accepted approach (Reference A.1-5) of correlation is assumed, the conditional core damage frequency given a seismic event will remain unaltered whether equipment is out of service or not. Thus, the risk increase due to out of service equipment cannot be greater than the total SCDF estimated by the bounding method used in Reference A.1-5. That is, for the FNP site, the delta SCDF from equipment out of service cannot be greater than 4.51E-6/yr.

Evaluate Bounding LERF Contribution

A review of plant specific and generic information on LERF contributors for internal events and seismic events was performed. For internal events, LERF is typically associated with the Interfacing System Loss of Coolant Accidents (ISLOCAs), Steam Generator Tube Ruptures (SGTR), and failures of containment isolation. Based on several recent PWR SPRAs, the tubes of the steam generators are judged to not be vulnerable to seismic events based on their ductile materials. Also, ISLOCA has not been found to be a significant contributor to LERF for SPRAs. That is, the usual failures leading to ISLOCA are failures of valve and check valve internals. For seismic events, valves are found to have high seismic capacity, so these failure modes are not contributors to seismic LERF. However, failure of containment isolation has been identified by SPRAs as a contributor to seismic LERF.

Seismic PRAs have generally found that structural failures and failure of containment isolation are the significant contributors to seismic LERF. At FNP, the Category I structures have high seismic capacity based on the initial work for the future SPRA. Therefore, seismic failure of containment isolation is judged to be the most significant contributor to SLERF.

While seismic failures of containment isolation have some degree of correlation with seismic CDF failures, a large majority of the potential failures, such as valves, would be significantly uncorrelated with the dominant seismic CDF failures. Therefore, the seismic fragility for containment isolation failure (which is based on the same conservative HCLPF for CDF) can be convolved with the seismic CDF fragility in order to estimate the seismic LERF. That is, the SLERF is estimated by the convolution of the seismic hazard with the core damage fragility and the LERF fragility, for each of the four spectral frequencies noted in the SCDF discussion, and averaged over the resulting values. Using the above inputs, the total estimated FNP SLERF is determined to be 2.07E-6 (Reference A.1-7).

Therefore, a RICT penalty of 2.07E-6 will be used as the bounding estimate of SLERF ($ILERF_{seismic}$) for the LAR submittal RICT calculations.

Conclusion

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

The RICT and RMAT calculations are based on the technical basis provided above.

The actual RICT and RMAT calculations performed by the CRMP tool are based on adding an incremental 4.51E-6/year and 2.07E-6/year seismic contribution to the configuration-specific delta CDF/delta LERF attributed to internal and fire events contributions. Thus, any change in risk due to the seismic contribution from the un-modelled seismic scenarios is accounted for by adding a permanent seismic contribution of 4.51E-6/2.07E-6 to the CRMP logic model that is used to quantify instantaneous CDF/LERF whenever a RICT is in effect. This method ensures that an incremental seismic CDF/LERF equal to the bounding SCDF/SLERF is added to internal and fire events incremental CDF/LERF contribution for every RICT occurrence.

The ICDP/ILERP acceptance criteria of 1E-5/1E-6 are used within the CRMP framework to calculate the resulting RICT and RMAT based on the total configuration-specific delta CDF/LERF accounting for internal events, fire and seismic CDF/LERF contributions.

References

A.1-1. Alabama Power Company, “Farley Nuclear Plant, Units 1 and 2, Individual Plant Examination of External Events,” June 1995.

A.1-2. Lettis Consultants International (LCI), Inc., “Farley Seismic Hazard and Screening Report,” LCI Project 1041, Rev. 1, October 30, 2013.

A.1-3 Alabama Power Company, “Joseph M. Farley Nuclear Plant - Units 1 and 2 Seismic Hazard and Screening Report for CEUS Sites,” NL-14-0342, March 31, 2014.

A.1-4. Generic Issue 199, “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,” IN2010-18, September 2, 2010.

A.1-5. Staff Report, “Implications of Updated Probabilistic Seismic Hazard Estimates In Central And Eastern United States On Existing Plants, Safety/Risk Assessment,” ML100270639, August 2010.

A.1-6. Southern Nuclear Co., “FNP Level 1 and 2 PRA Model Revision 9 - at power, internal events,” Calculation No. PRA-BC-F-14-001 Farley IE Model Rev 9 Ver 3, January 9, 2014.

A.1-7 Southern Nuclear Co., “Seismic Risk Evaluation based on IPEEE and EPRI 2014 Farley Seismic Hazard,” PRA-BC-F-17-002, September 28, 2017.

Attachment 2: Evaluation of External Event Challenges and IPEEE Update Results

As shown in Figure E3.1, there are three parts to the process for addressing external hazards for the RICT Program. Step 2 of the process addresses beyond design basis hazards that were not screened out in Step 1. As shown in Enclosure 3 Attachment 1, a bounding analysis approach was used to address the impact of seismic risk on RICT Program calculations. As such, the primary purpose of this attachment is to address Step 3 of the process from Figure E3.1.

As described in this enclosure, the incremental risk associated with challenges to the facility that do not exceed the design capacity must be accounted for. This attachment also provides results of the hazard screening performed as part of Step 1 of the process from Figure E3.1. Seismic is the only hazard that was not screened out from Step 2 in Step 1.

Step 1 Hazard Screening Except Seismic Events

The FNP IPEEE for Units 1 and 2 (Reference. A.2-1) provides an assessment of the vulnerability of the site to these hazards. The FNP IPEEE external hazard screening evaluation was updated to support this LAR. The updated evaluation of other external hazards for FNP Units 1 and 2 (Reference A.2-2) provides an assessment of the vulnerability of the site to these hazards. In general, the FNP site screened these external hazards based on Table 10-1 of NUREG/CR-2300, PRA Procedures Guide (Reference A.2-13), and performed a bounding evaluation (Reference A.2-2) for those hazards not subject to screening (aircraft impact, extreme winds and tornadoes, external flooding including intense local precipitation, industrial and military facility accidents, pipeline accidents, transportation accidents, and turbine-generated missiles). The bounding evaluation determined that the bounding CDF for beyond design basis conditions is less than 1E-6 per year. Table E3.A2.1 presents the results of the updated IPEEE analysis.

This screening and bounding evaluation assures that safety related equipment is not affected from beyond design basis events other than seismically induced impacts, which are evaluated in Attachment 1 of this Enclosure.

Step 2 Risks from Hazard Challenges Except Seismic Events

Table E3.A2.1 reviews the bases for the evaluation of these hazards, identifies any challenges posed, and identifies any additional treatment of these challenges, if required. The conclusions of the assessment, as documented in Table E3.A2.1, assures that the hazard either does not present a design-basis challenge to FNP, or is adequately addressed in the PRA.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
Seismic Events	Seismic events treated using a bounding approach with change to RICT Program criteria (see Attachment 1 to this enclosure).	Seismically induced loss of offsite power (LOSP) is a challenge within the design basis.	Addressed as part of internal events treatment of LOSP.
Accidental Aircraft Impacts	There are no airports within 10 miles of the plant. There are no military facilities or military training routes close to the plant. Aircraft hazard is not a design basis hazard event for the plant and the UFSAR (Reference A.2-3) using the most recent data confirms this conclusion. As a result, beyond design basis challenges from accidental aircraft impacts are screened out. (Reference A.2-2)	Aircraft impact induced LOSP is a potential challenge within the design basis.	Projected air traffic does not pose a credible challenge to FNP. The likelihood of damage causing a LOSP is judged to be sufficiently small that it will not significantly impact the RICT Program calculations and it can be excluded from RICT Program evaluation.
Avalanche	Topography is such that no avalanche is possible as plant is not located near large mountains where snow avalanches are prevalent.	Impact of cascade of snow or rock would be damage to the exterior structure	The effect of an avalanche does not pose a credible risk to FNP.
Biological Event	The accumulation or deposition of vegetation or organisms (e.g. zebra mussels, clams, fish) on an intake structure or internal to a system that uses an intake structure would not occur as the Chattahoochee River is not the Ultimate Heat Sink (UHS) for FNP. The Service Water Storage Pond provides this service. As this is slow to develop, there would be adequate warning for these events.	There are no challenges presented to the FNP site from biological events.	Excluded from RICT Program evaluation.
Coastal Erosion	FNP is a riverine site located inland.	There are no challenges presented to the FNP site from coastal erosion.	Excluded from RICT Program evaluation.
Drought	Drought is a slowly developing hazard. The plant location (riverine site with upstream dams: Walter F. George Dam and Columbia Lock and Dam; and downstream	There are no challenges presented to the FNP site from coastal erosion.	Excluded from RICT Program evaluation.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program			
External Hazard	**Current Risk Basis**	**Challenge(s) Posed**	**Disposition for RICT Program**
	dam, Jim Woodruff Dam) precludes impact on FNP.		
External Flooding	The external flooding hazard at the site was recently updated as a result of the post-Fukushima 50.54(f) Request for Information. The flood hazard reevaluation report (FHRR) was submitted to NRC for review on October 20, 2015 (Reference A.2-4). The NRC concluded in (Reference A.2-5) that the reevaluated flood hazards information in (Reference A.2-4) is suitable for the assessment of mitigating strategies (i.e., defines the mitigating strategies flood hazard information described in guidance documents currently being finalized by the industry and NRC staff) for Farley. Further, the NRC staff has concluded that the reevaluated flood hazard information is a suitable input for other assessments associated with Near-Term Task Force Recommendation 2.1 "Flooding." The results in (Reference A.2-4) indicate that the frequency of a local intense precipitation (LIP) event capable of producing flood magnitudes reported in the FHRR is estimated to be well below 10^{-6}/yr. The second mechanism evaluated in the FHRR is combined events river flooding that is primarily caused by a probable maximum precipitation (PMP) event and wind-wave action. However, this mechanism is estimated to produce a maximum flood elevation that will not top the vehicle barrier system (VBS) surrounding the site. Although wind-wave action may produce sloshing over the VBS, the volume expected due to sloshing will	Weather induced Loss of Offsite Power (LOSP) is a potential challenge, e.g., Flood induced loss of Emergency AC power, Aux. Feedwater (TDAFW Pump), Low Pressure/Decay Heat Removal pumps, and High Pressure/Makeup Pumps.	The combined effects of river flooding will not challenge the plant due to the VBS and site grade. LIP will be addressed by several modifications as a result of the Mitigating Strategies Assessment (MSA) in response to NRC Order EA-12-049, "Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (ADAMS Accession No. ML 12054A735). These modifications include building protection curbs around several key doors to keep flood waters from entering the building housing Key SSCs and FLEX equipment. Therefore, FLEX will be able to cope with the reevaluated flood hazard, and the modifications will provide protection for Key SSCs to maintain Key Safety Functions (KSFs) throughout the flooding event. Given the extremely low likelihood of an LIP event and the ability of FLEX to cope with the reevaluated flood hazard, the risk from an LIP event is sufficiently low enough to not warrant further analysis.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program			
External Hazard	**Current Risk Basis**	**Challenge(s) Posed**	**Disposition for RICT Program**
	not challenge site grade or any plant SSCs.		
Extreme Winds and Tornados (including generated missiles)	FNP has been designed for extreme winds and tornado loadings that are substantially higher than the design basis events presently required. Most of the safety related structures, systems and components (SSCs) are protected from tornado missiles using barriers with thicknesses exceeding the current requirements based on recent tornado hazard analysis. Detailed tornado missile risk analysis has shown that the frequency of missile damage to target groups is less than $7x10^{-7}$ per year per unit, which is less than the screening criterion of 1E-6 per year (Reference A.2-2). On that basis, beyond design basis challenges from extreme winds & tornados are screened out.	Loss of offsite power from extreme winds & tornados is a potential challenge within the design basis. The site is currently evaluating tornado missiles in response to RIS 15-06. Tornado missile protection (TMP) vulnerabilities are in the process of being evaluated.	Weather-related LOSP and recovery are included in data used for internal events PRA (Reference A.2-6). No further analysis required. Results of the TMP evaluation will be reflected in the extreme winds and tornados screening evaluation.
Fog	Water droplets suspended in the atmosphere at or near the Earth's surface that limit visibility affect the frequency of occurrence of other hazards (e.g. highway accidents, aircraft landing and take-off accidents) and is indirectly considered. Fog has a rare occurrence in the site region. Section 2.3.2.2 of UFSAR states that visibility of less than 1/4 mile occurs less than 1.3 percent of the time.	There are no challenges presented to the FNP site from fog.	Excluded from RICT Program evaluation.
Frost	Snow and Ice govern this risk.	There are no challenges presented to the FNP site from frost.	Excluded from RICT Program evaluation.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program			
External Hazard	**Current Risk Basis**	**Challenge(s) Posed**	**Disposition for RICT Program**
Hail	Showery precipitation in the form of irregular pellets or balls of ice may occur. Hail may occur but there are no openings in the walls or roofs of safety related buildings through which hail may enter and damage essential equipment. Tornado missile protection features, structural walls and roods are adequate to withstand the impact of hail.	There are no challenges presented to the FNP site from hail.	Excluded from RICT Program evaluation.
High Summer Temperature	The highest recorded temperature at Dothan Airport was 108°F. The HVAC systems are designed to maintain prescribed building temperatures during outside temperature variations between 20°F and 95°F. Even if the maximum temperature exceeds the design limits for HVAC systems, such exceedance lasts only for a brief period and, given the thermal inertia of the concrete structures where safety-related equipment are located, will not have any impact.	There are no challenges presented to the FNP site from high summer temperature.	Excluded from RICT Program evaluation.
High Tide, Lake Level or River Stage	This event is of negligible impact on plant. The plant location (riverine with upstream and downstream dams) preclude impact on plant due to this hazard. See External Flooding discussion for more information.	There are no challenges presented to the FNP site from high tide, lake level, or river stage.	Excluded from RICT Program evaluation.
Hurricane	FNP is not on the coast and hurricane wind effects are bounded by extreme winds and tornados assessment.	See extreme winds and tornados.	See extreme winds and tornados.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program			
External Hazard	**Current Risk Basis**	**Challenge(s) Posed**	**Disposition for RICT Program**
Ice Cover	Accumulation of frozen water on bodies of water (e.g. rivers) or on structures, systems and components. Icing does not normally occur on the Chattahoochee River at FNP. The only incidence of icing occurred in 1961 along the banks in slack water areas. No record of the river being iced over at this location has been found. Therefore, there would be no interference with the flow of water into the river water intake due to ice. Even if the surface did become frozen there would be no interference with withdrawal of water by the river water intake due to depth of water in the river (UFSAR Section 2.4.7).	There are no challenges presented to the FNP site from ice cover.	Excluded from RICT Program evaluation.
Turbine-Generated Missiles	The probabilistic analysis performed for failures of turbines in Units 1 & 2 shows the probability of turbine missile damage is less than the NRC accepted value (per RG 1.115, Reference A.2-12) of 1x10-7 per year. To further reduce the probability of turbine failure, FNP has adopted a rigorous maintenance program. Therefore, given the worst case probability of turbine missile damage of 1E-7 the bounding CDF assuming a CCDP of 1.0 is less than 1E-6 per year. (Reference A.2-2) Beyond design basis challenges from turbine-generated missiles are screened out.	Loss of offsite power from turbine missiles is a potential challenge within the design basis.	The likelihood of damage causing a LOSP is judged to be sufficiently small that it will not significantly impact the RICT Program calculations and it can be excluded from RICT Program evaluation.
Internal Fires	FNP Internal Fire model addresses risk from internal fires.	Internal Fire impacts are evaluated in the internal Fire PRA.	Internal Fire impacts are evaluated in the internal Fire PRA.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program			
External Hazard	**Current Risk Basis**	**Challenge(s) Posed**	**Disposition for RICT Program**
Internal Flooding	FNP Internal events and internal flooding model addresses risk from internal flooding events.	Internal Flooding impacts are evaluated in the internal flooding PRA.	Internal Flooding impacts are evaluated in the internal flooding PRA.
Landslide	FNP's location prevents landslides from occurring as there are no steep hills.	There are no challenges presented to the FNP site from landslide.	Excluded from RICT Program evaluation.
Lightning	Lightning strikes are not uncommon in nuclear plant experience. They can result in losses of off-site power or surges in instrumentation output if grounding is not fully effective. The latter events often lead to reactor trips. This was considered in plant design.	There are no challenges presented to the FNP site from lightning.	Excluded from RICT Program evaluation.
Low Lake Level or River Stage	A decrease in the water level of the lake or river does not impact FNP. A decrease in the water level of the lake or river does not impact FNP as FNP does not rely on Chattahoochee River for the UHS since the storage pond provides the necessary UHS requirements.	There are no challenges presented to the FNP site from low lake level or river stage.	Excluded from RICT Program evaluation.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program			
External Hazard	**Current Risk Basis**	**Challenge(s) Posed**	**Disposition for RICT Program**
Low Winter Temperature	The lowest recorded temperature at Dothan Airport was 5°F; the plant design basis is 17°F. The HVAC systems are designed to maintain prescribed building temperatures during outside temperature variations between 20°F and 95°F. Even if the minimum temperature exceeds the design limits for HVAC systems, such exceedance lasts only for a brief period and, given the thermal inertia of the concrete structures where safety-related equipment are located, will not have any impact. Therefore, the temperatures inside the plant buildings are expected to be higher than 17°F.	There are no challenges presented to the FNP site from low winter temperature.	Excluded from RICT Program evaluation.
Meteorite or Satellite Impact	A meteoroid or artificial satellite that releases energy due to its disintegration in the atmosphere above the Earth's surface, direct impact with the Earth's surface, or a combination of these effects. This hazard is of negligible likelihood of impact to the site (very low event probability).	There are no challenges presented to the FNP site from meteorite or satellite impact.	Excluded from RICT Program evaluation.
Forest or Range Fires	Fires at nearby facilities, onsite chemical storage, nearby transportation routes, or pipelines are addressed within those external hazard categories. For forest fires, UFSAR Sec 2.3.6 (Reference A.2-3) states that wooded areas are sufficiently far from the plant structures that brush and forest fires do not present a hazard. (Reference A.2-2)	There are no challenges presented to the FNP site from forest fires.	Excluded from RICT Program evaluation.
Industrial or Military Facility Accident	No military bases or firing ranges, oil pipelines, or tank farms are located within a 10-mile radius of the plant site. Therefore, the hazards from industrial and military facility accidents are screened out from FNP PRA. (Reference A.2-2)	There are no challenges presented to the FNP site from accidents at nearby facilities.	Excluded from RICT Program evaluation.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
Release of Chemicals in Onsite Storage	Chemicals stored near FNP have been evaluated annually since the OL issuance (Reference A.2-2). Procedures are in place to assess the impact of any new chemical procured for plant operations on control room habitability based on the toxicity limits given in RG 1.78 (Reference A.2-7). Based on the evaluations reported in the UFSAR (Reference A.2-3) on storage and handling of toxic chemicals near the site, this hazard group does not pose a credible threat to FNP Units 1 & 2. (Reference A.2-2)	There are no challenges presented to the FNP site from chemicals stored onsite.	Excluded from RICT Program evaluation.
River Diversion	UFSAR Section 2.4.9 states that the river upstream from the site does not have sufficiently high banks to cause a potential diversion of the river and bypass of the intake structure. With Lake Seminole varying between el 76 ft MSL and 78 ft MSL, a temporary blockage of the river upstream from FNP would not seriously affect the quantity of water available to the river water intake. Even if the river was temporarily blocked, cooling water could be obtained from the storage pond.	There are no challenges presented to the FNP site from river diversion.	Excluded from RICT Program evaluation.
Sand or Dust Storm	A strong wind storm with airborne particles of sand and dust is not relevant for this region.	There are no challenges presented to the FNP site from sand or dust storms	Excluded from RICT Program evaluation.
Seiche	This is an oscillation of the surface of a landlocked body of water that can vary in period from minutes to several hours; however, there is no large body of water close to the site for this event.	There are no challenges presented to the FNP site from seiche.	Excluded from RICT Program evaluation.
Snow	The 100 year snow load is estimated as 10 psf. The design basis roof live load for seismic Category I structures is at least 20 psf.	There are no challenges presented to the FNP site from snow.	Excluded from RICT Program evaluation.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
Soil Shrink-Swell Consolidation	The relative change in volume of the soil as a result of the type of soil and the amount of moisture. This is slow to develop and procedures are in place to monitor differential settlement (UFSAR Section 2B.7.3.1)	There are no challenges presented to the FNP site from soil shrink-swell consolidation.	Excluded from RICT Program evaluation.
Storm Surge	FNP is located inland and is not affected by storm surge.	There are no challenges presented to the FNP site from storm surge.	Excluded from RICT Program evaluation.
Toxic Gas	Toxic gas is covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident	Toxic gas is covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident	Toxic gas is covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident
Transportation Accidents	Analysis of postulated accidents on nearby transportation routes has shown (Reference A.2-2) that they do not pose a credible threat to FNP since these routes are farther than the safe distances specified in RG 1.78 (Reference A.2-7) and RG 1.91 (Reference A.2-8).	There are no challenges presented to the FNP site from transportation accidents.	Excluded from RICT Program evaluation.
Pipeline Accidents (e.g., natural gas)	A 6-in gas pipeline passes about 2.5 miles east of the main plant building. This is a grade B pipe with a nominal wall thickness of 0.188 in. and an average depth of 30 in. It carries 12 million cubic feet per day. In Section 2.2.3.2 of the UFSAR (Reference A.2-3), it is stated that an explosion or fire following a break of this pipe would not be hazardous for FNP. Therefore, the hazard posed by pipeline accidents is screened out from the FNP PRA. (Reference A.2-2) Beyond design basis challenges from pipeline accidents screened out.	Loss of offsite power from blast pressure damage to SSCs from pipeline accidents is a potential challenge within the design basis.	Based on the UFSAR evaluation, the pipeline does not pose a challenge to FNP. As a result, the likelihood of damage causing a LOSP is judged to be sufficiently small that it will not significantly impact the RICT Program calculations and it can be excluded from RICT Program evaluation.

Table E3.A2.1 Evaluation of Hazard Challenges and Disposition for RICT Program

External Hazard	Current Risk Basis	Challenge(s) Posed	Disposition for RICT Program
Tsunami	FNP is located inland is not exposed to the Tsunami threat	There are no challenges presented to the FNP site from tsunamis.	Excluded from RICT Program evaluation.
Volcanic Activity	Not applicable to the site because of location (no active or dormant volcanoes located near plant site)	There are no challenges presented to the FNP site from volcanic activity.	Excluded from RICT Program evaluation.
Waves	FNP is located inland and is not affected by any wave activity.	There are no challenges presented to the FNP site from waves.	Excluded from RICT Program evaluation.

Step 3 Seismic-Induced LOSP Challenges

For the FNP site, the only incremental risk associated with challenges to the facility that do not exceed the design capacity, which is not already addressed, is the seismically-induced LOSP. The methodology for computing the seismically-induced LOSP frequency is simply a convolution of the mean seismic hazard curve and the offsite power fragility. The Farley seismic hazard curve is the re-evaluated hazard submitted to NRC (Reference A.2-9) in response to the 50.54(f) request regarding Recommendation 2.1 of the NRC Fukushima Near Term Task Force.

Table E3.A2.2 provides the mean seismic hazard, represented by a series of discrete seismic hazard intervals from just below the FNP operating basis earthquake to significantly above the safe shutdown earthquake, and the LOSP failure probability for each seismic interval based on the fragility of offsite power, represented by failure of ceramic insulators in the offsite power switchyard. The failure probabilities are based on the fragility data from Table 4B-1 of the RASP Handbook (Reference A.2-10):

Median Offsite Power Capacity = 0.3g PGA, β_R = 0.3, β_U = 0.45

Given the mean frequency and failure probability for each seismic interval, it is straightforward to compute the estimated frequency of seismically induced loss of offsite power for the FNP site by taking the product of the interval frequency and the offsite power failure probability. As shown in Table E3.A2.2, the total seismic LOSP frequency is the sum of interval frequencies, or approximately 5E-6/yr.

Table E3.A2.2
Seismic LOSP Frequency Based on FNP Seismic Hazard and RASP Handbook Fragility Data (Reference A.2-10)

Seismic Interval (g)	Represent ative Acceler- ation (g)	Interval Frequency (/yr)	Offsite Power Failure Prob.	Weighted Average LOSP freq
0.05 - 0.1	0.07	1.13E-04	3.77E-03	4.26E-07
0.1 - 0.3	0.17	2.11E-05	1.55E-01	3.27E-06
0.3 - 0.5	0.39	9.35E-07	6.82E-01	6.37E-07
0.5 - 0.7	0.59	1.80E-07	8.95E-01	1.61E-07
0.7 - 0.9	0.79	5.56E-08	9.64E-01	5.36E-08
0.9 - 1.1	0.99	2.25E-08	9.87E-01	2.22E-08
1.1 - 1.3	1.20	1.31E-08	9.95E-01	1.30E-08
1.3 - 1.5	1.40	2.39E-09	9.98E-01	2.38E-09
>1.5	2.12	6.40E-09	1.00E+00	6.40E-09
Total Seismic LOSP Frequency =				**4.59E-06**

The internal events PRA relies on the loss of offsite power data in Reference A.2-11. Based on the FNP internal events PRA (Reference A.2-4), the total LOSP frequency is approximately 2E-2/yr. from plant-centered, grid-related, and weather-related causes. Applying the non-recovery probability at 24 hours to each of these causes of LOSP results in a frequency of unrecovered loss of offsite power of 1.5E-3/yr. that is already included in the internal events PRA.

The seismically-induced (unrecoverable) LOSP frequency (5E-6/yr) is therefore less than 1% of the total unrecovered LOSP frequency. This frequency is judged to be a sufficiently small fraction that it will not significantly impact the RICT Program calculations and it can be omitted.

Conclusions

Based on this analysis of external hazards for FNP Units 1 and 2, no additional external hazards need to be added to the existing PRA model. The evaluation concluded that the hazards either do not present a design-basis challenge to FNP, the challenge is adequately addressed in the PRA, or the hazard has a negligible impact on the calculated RICT and can be excluded.

References

A.2-1. Alabama Power Company, “Farley Nuclear Plant, Units 1 and 2, Individual Plant Examination of External Events,” June 1995.

A.2-2. “Joseph M. Farley Nuclear Plant Units 1 and 2, Evaluation of Other External Hazards,” Southern Nuclear PRA Report, Revision 0, December 31, 2013.

A.2-3 Joseph M. Farley Nuclear Plant Units 1 and 2 UFSAR, Rev 28 December 2017

A.2-4. Joseph M. Farley Nuclear Power Plant – Units 1 & 2 Flood Hazard Reevaluation Report (FHRR), Version 1.0, NRC Docket No. 50-348 & 50-364, October 20, 2015

A.2-5 NRC ADAMS Accession No. ML15343A418 – “Joseph M. Farley Nuclear Plant, Units 1 and 2 – Interim Staff Response to Reevaluated Flood Hazards Submitted in Response to 10 CFR 50.54(f) Information Request – Flood-Causing Mechanism Reevaluation (TAC No. MF7039 and MF7040),” December 10, 2015

A.2-6 “FNP Level 1 and 2 PRA Model Revision 9 - at power, internal events,” PRA-BC-F-14-001 Farley IE Model Rev 9 Ver 3, January 28, 2014.

A.2-7 Regulatory Guide 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release,” U.S. Nuclear Regulatory Commission, Revision 1, 2001

A.2-8 Regulatory Guide 1.91, “Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Revision 1, February 1978

A.2-9 Alabama Power Company, “Joseph M. Farley Nuclear Plant - Units 1 and 2 Seismic Hazard and Screening Report for CEUS Sites,” NL-14-0342, March 31, 2014.

A.2-10 “Risk Assessment of Operational Events Handbook, Volume 2 – External Events,” Revision 1.01, U.S. Nuclear Regulatory Commission, January 2008.

A.2-11 "Losses of Off-Site Power at U.S. Nuclear Power Plants – Through 2001, Final Report," EPRI TR-1002987, Electric Power Research Institute, April 2002.

A.2-12 Regulatory Guide 1.115, "Protection Against Turbine Missiles," U.S. Nuclear Regulatory Commission, Revision 2, January 2012.

A.2-13 ANS-IPEEE-NRC, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments of Nuclear Power Plants," Report NUREG/CR-2300, 1983.

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, “Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines”

Enclosure 4

Baseline CDF and LERF

Table of Contents

1.0 Introduction

The purpose of this enclosure is to demonstrate that the total Core Damage Frequency (CDF) and total Large Early Release Frequency (LERF) are below the limits established in Regulatory Guide (RG) 1.174 (Reference 1), which are 1E-4/year for CDF and 1E-5/year for LERF. These limits allow for the risk metrics of NEI 06-09 (Reference 2) to be applied to the Farley Nuclear Plant (FNP) Risk Informed Completion Time (RICT) Program.

Table E4.1 reflects the Unit 1 and Unit 2 CDF and LERF values that resulted from a quantification of the baseline internal events (including internal flooding) (References 4 and 5) and fire Probabilistic Risk Assessment (PRA) average annual models (Reference 3). Table E4.1 also includes the seismic CDF/LERF values (Reference 6). Other external hazards, as discussed in Enclosure 3, are below accepted screening criteria and therefore do not contribute significantly to the totals. The values for the internal events and fire PRAs represent the average of Train A and Train B plant configuration alignments CDF/LERF results for each unit.

Table E4.1
Total Baseline Average Annual CDF/ LERF

Farley Unit 1		
Source	Baseline CDF/year	Baseline LERF/year
Internal Events PRA	8.91E-06	1.28E-07
Fire PRA	8.35E-05	4.21E-06
Seismic	4.51E-06	2.07E-06
Other External Events	Screened out	
TOTAL UNIT 1	9.69E-05	6.41E-06

Farley Unit 2		
Source	Baseline CDF/year	Baseline LERF/year
Internal Events PRA	8.76E-06	1.03E-07
Fire PRA	7.89E-05	4.51E-06
Seismic	4.51E-06	2.07E-06
Other External Events	Screened out	
TOTAL UNIT 2	9.22E-05	6.68E-06

As demonstrated in Table E4.1, the total CDF and total LERF for each unit are within the limits set forth in RG 1.174, which permit small changes in risk that may occur during entries into the RICT Program. Therefore, the FNP RICT Program is consistent with NEI 06-09 guidance.

The values shown in Table E4.1 are a snap shot in time (Reference 3) and are subject to change based on the on-record PRA models that support the RICT Program. The RICT Program will monitor these values to ensure that annual average CDF and LERF are reasonably within RG 1.174 limits of 1E-04 and 1E-05 as a condition of program implementation requirement. Enclosure 9 provides additional information on the RICT Program monitoring process.

2.0 References

1. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing basis," May 2011 (ADAMS Accession No. ML090410014).

2. NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Nuclear Energy Institute, Revision 0-A, October 2012 (ADAMS Accession No. ML 122860402).

3. SNC Calculation F-RIE-FIREPRA-U00-014, "Fire Probabilistic Risk Assessment Quantification and Summary Report", Revision 1

4. SNC Calculation F-RIE-IEIF-U01,"FNP Unit 1 Internal Event, Internal Flooding and LERF PRA Model Update" Rev. 10, Ver. 2.0

5. SNC Calculation F-RIE-IEIF-U02, "FNP Unit 2 Internal Event, Internal Flooding and LERF PRA Model Update", Rev. 10, Ver. 2.0

6. SNC Calculation PRA-BC-F-17-002, "Seismic Risk Evaluation based on IPEEE and EPRI 2014 Farley Seismic Hazard", Version1

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines"

Enclosure 5

PRA Model Update Process

Table of Contents

1.0 Introduction

The administrative controls applicable to the Probabilistic Risk Assessment (PRA) models used to support the Risk Informed Completion Time (RICT) Program ensure that these models reflect the as-built, as-operated plant. Plant changes, including physical modifications and procedure or operating practice changes, are reviewed prior to implementation to determine if they could impact the PRA models. If so, the process then determines the quantitative significance of the change and, if appropriate, implements the PRA model change concurrently with the plant change. If the change is not quantitatively significant, the PRA model change is prioritized for implementation at a routine model update. Such pending changes are considered when evaluating other changes until they are fully implemented into the PRA models. Routine updates are performed, as a minimum, every two fuel cycles. If a quantitatively significant change cannot be implemented in the PRA model such that it could adversely affect RICT calculations, alternatives including bounding analyses or restrictions on the use of the RICT program are put in place until the PRA model can be changed.

2.0 PRA Model Update Process

2.1 Internal and Fire Events PRA Maintenance and Update

The Southern Nuclear Operating Company (SNC) risk management process ensures that the applicable PRA model reflects the as-built and as-operated plant for each of the Farley Nuclear Plant (FNP) units, as required by Regulatory Guide 1.200 (Reference 2). The process delineates the responsibilities and guidelines for updating the full power internal events and internal fire PRA models at all operating SNC sites, and it includes both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential impact areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience) and assessing the risk impact of unincorporated changes. The process also provides for controlling the model and associated electronic files.

The SNC PRA update process procedures include a requirement to maintain the total CDF and LERF mean values from all quantified sources documented in the LAR, including impact of changes to fire ignition frequency updates, within reasonable limits of the RG 1.174 risk acceptance guidelines of 1E-4/yr. (CDF) and 1E-5/yr. (LERF) (Reference 3).

2.2 Review of Plant Changes for Incorporation into the PRA Model

1. Plant Changes (including both physical modifications to the facility and changes to procedures or operating practices) are reviewed as follows:

    a. Modifications to the physical plant are reviewed for changes to maintain the PRA consistent with the as-designed plant. The review of design changes, e.g., Design Change Packages (DCP), Minor Design Changes (MDC), etc., is performed on an on-going basis. All design changes expected to impact or result in a need to change the baseline PRA model are identified in the PRA change log.

    b. Modifications to plant procedures, Technical Specifications, and other licensing documents are reviewed to maintain the PRA consistent with the as-operated plant. The review is performed on an on-going basis. Licensing Document Change Requests (LDCR) expected to significantly impact or change the baseline PRA model are identified in the PRA model change log.
    c. Reliability data, unavailability data, initiating events frequency data, human reliability data, and other such PRA inputs are reviewed at least every two fuel cycles to maintain the PRA consistent with the as-operated plant.

2. If a quantitatively significant change to the PRA model is identified, it is accounted for in the model prior to the implementation of that plant change, including a physical modification, a procedure change, or other changes as noted in Item (1).

3. Following the data review performed at least every two fuel cycles, the PRA is reviewed to account for cumulative changes identified by the analysis.

4. If PRA model errors are discovered, they are reviewed to determine the quantitative impact on PRA results. Errors that result in quantitatively significant changes to the PRA model are corrected as soon as possible. Other errors are corrected on a completion schedule that is determined based on their priority.

5. When a PRA model change is required but cannot be immediately implemented for a quantitatively significant plant change or model error, the process calls for either one of the following actions:

   a. Alternative analyses to conservatively bound the expected risk impacts of changes on the model are 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. The use of such bounding analyses is consistent with NEI 06-09 (Reference 1).

   b. Appropriate administrative restrictions on the use of the RICT Program for extended CTs are put in place until the model changes are completed.

3.0 References

1. NEI 06-09, “Risk-Informed Technical Specifications Initiative 4B: Risk-Managed Technical Specifications (RMTS) Guidelines,” Nuclear Energy Institute, Revision 0-A, October 2012 (ADAMS Accession No. ML 122860402).

2. Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities,” US Nuclear Regulatory Commission, Revision 2, March 2009.

3. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific changes to the Licensing Basis,” May 2011 (ADAMS Accession No. ML12321A054)

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, “Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines”

Enclosure 6

Attributes of the CRMP Model

Table of Contents

1.0 Introduction

This enclosure describes the process for adapting the peer-reviewed baseline Probabilistic Risk Assessment (PRA) models for use in the Configuration Risk Management Program (CRMP) software to support the Risk Informed Completion Time (RICT) Program. Farley Nuclear Plant (FNP) intends to employ a CRMP software tool which provides for real time recalculation of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) for configuration risk. The baseline PRA models are separate internal events (including internal flooding) and internal fires models, which calculate average annual risk. The CRMP model used in the RICT Program must integrate results for all modeled hazard groups and determine CDF and LERF for actual plant conditions which exist at the time. The process employed to adapt the baseline models for CRMP use is demonstrated 1) to preserve the CDF and LERF quantitative results, 2) to maintain the quality of the peer-reviewed PRA models and 3) to correctly accommodate changes in risk as required due to time-of-year, time-of-cycle and configuration-specific considerations as required. As indicated in Enclosure 1, the representative RICT values reported in Enclosure 1 are calculated using separate zero-maintenance annual average PRA models which include the internal events (including internal flooding) PRA model, internal fire events PRA model that reflect NFPA-805 implemented plant modifications, seismic bounding delta CDF/LERF values, and main control room (MCR) abandonment bounding delta CDF/LERF values. The Farley Maintenance Rule a(4) CRMP model that reflects the as-built and as-operated plant condition including credit for NFPA-805 modifications will be modified (similar in scope to the currently implemented VEGP 4B CRMP model) prior to implementation of the Farley RICT Program. Quality controls and training programs applicable for the CRMP tool are also discussed in this enclosure. The MCR abandonment bounding delta CDF/LERF values are subject to replacement either with updated values as the model is updated or optionally reflected as a logic change within the Internal Fire PRA model. The Seismic bounding delta CDF/LERF values are subject to replacement with updated values as the model is update

2.0 Process

The baseline PRA models for internal events (including internal flooding) and internal fires are peer-reviewed models, updated to incorporate resolution of relevant peer review findings and to incorporate plant changes that reflect the as-built and as-operated plant. These models will then be modified using the following process to create a single top CRMP model, which also includes changes needed to facilitate configuration-specific risk calculations.

Each step in the process is documented using, as required, separate reports or calculations, which provide for necessary reviews and approvals of the changes being applied. The significant steps of the process are described below:

Step 1: This step represents the model for internal events and internal flooding which was subjected to the peer review process.

Step 2: This step represents the model for internal fires which was subjected to the peer review process.

Step 3: This step represents the modification of the internal events and internal flooding model to resolve peer review findings determined to be relevant to the use of the models in the RICT Program, as well as updates to address plant changes.

Step 4: This step represents the modification of the internal fires model to resolve peer review findings determined to be relevant to the use of the models in the RICT Program, as well as updates to address plant changes.

Step 5: This step makes changes to the internal events (including internal flooding) and fire PRA models to include systems, structures, and components (SSCs) that are in the scope of the RICT Program but which are not part of the baseline PRA models. An evaluation of the RICT Program scope against the baseline PRA model scope is performed to identify SSCs which are not part of the baseline model and which need to be included to support configuration risk evaluations for LCOs in the scope of the RICT Program. It is expected that future revisions to the baseline PRA model will incorporate those SSCs that support configuration risk evaluations for the RICT program.

The changes being made to the existing baseline PRA models do not involve new methods; as such there is no need for any focused scope peer review. The associated LCOs are described in Enclosure 1.

Step 6: This step integrates the two baseline PRA models, following steps 4 and 5, into a single top fault tree model for calculation of CDF and LERF. The single top model is capable of evaluating the fire scenarios along with the internal events initiators and then combining the numerical results for use in the CRMP. At this step, the single top risk model calculates the total average annual CDF and LERF from internal events, internal floods, and internal fires.

Also at this step, the results obtained from the integrated model are validated against the baseline model results to ensure the single-top model is properly calculating CDF and LERF. The single top model accommodates such comparisons because it permits quantification of all initiating events, or a selection of initiating events, which facilitates comparisons to the two baseline PRA models.

At the completion of step 6, the two PRA baseline models are integrated, and the single top model is verified to provide quantitative results consistent with the two baseline models.

Step 7: This step optimizes, if required, the single top model to improve quantification time and is an intermediate step towards the next step.

At the conclusion of step 7, the quantified results from the optimized model are benchmarked to ensure the optimization process did not significantly alter the numerical results from the baseline PRA models.

Step 8: This step changes the model logic to account for variations in system success criteria based on the time of year or the time in the operating cycle as required. It also accounts for other specific changes needed to properly account for configuration-specific issues as required, which are either not evaluated in the baseline average annual model or are evaluated based on average conditions encountered during a typical operating cycle. The CRMP model used for the RICT Program is required to either conservatively model these variations or include the capability to account for the variations.

The types of changes implemented in the CRMP model are described in Table E6.2. Some specific examples of equipment alignment possibilities are shown (e.g., status of PORV block valves) but a number of other system alignments, such as high head charging and nuclear service water trains that are not shown but would be reflected in the CRMP model based on the configuration-specific equipment alignments in effect at the time of a RICT calculation.

Table E6.2
Changes Made During Translation to CRMP Model

Description	Basis for Change
Seismic Bounding Risk	Seismic risk is not included in the baseline PRA models. As justified in Enclosure 3 of this LAR, bounding seismic CDF and LERF values are calculated and included in the FNP baseline risk of the CRMP model.
Plant Availability (PAV) Event	The baseline PRA models account for the time the reactor operates at power by using a plant availability factor. This is appropriate for determining the average annual (time based) risk, but the factor is not applicable to configuration-specific risk calculated for the RICT Program. Therefore, the probability of the PAV event is set to 1.0 in the CRMP model. This change is necessary to adjust the modeled initiating event frequencies from a per year to per reactor year basis for use in the CRMP.

Table E6.2
Changes Made During Translation to CRMP Model

Description	Basis for Change
Maintenance Event Probabilities	Maintenance events in the baseline PRA models have probabilities based on the fraction of the year the equipment is unavailable. For the CRMP model, the actual configuration of equipment is known, so the maintenance event probabilities are set to 0. When components are in maintenance, these events (or equivalent events) are set to 1.
Primary Pressure Relief Control Interval for Anticipated Transient Without Trip (ATWT) Events	The FNP core design reflected in the baseline PRA model for ATWT events uses interval values to reflect impact of core life, whereas the CRMP model must reflect configuration-specific risk. Therefore the CRMP model is configured to select an interval value corresponding to the time in core life. The CRMP model will allow user input to select the appropriate time in life configuration applicable for RICT Program calculations.
PORV Block Valve Configuration	The success criteria in the baseline PRA for primary pressure relief during ATWT is based on average values for the period of time a PORV block valve is closed. The CRMP model must reflect configuration-specific risk. Therefore the CRMP model is configured to select a value of either zero or 1.0 for closure of the PORV block valves. The CRMP model will allow user input to select the appropriate configuration applicable for RICT Program calculations.

3.0 Administrative Controls

Departmental procedures and their sub-tier instructions and guidelines provide high level guidance and requirements for creating and maintaining the CRMP model for implementing the RICT Program at FNP. The procedures collectively implement the following requirements of NEI 06-09, Revision 0-A (Reference 1), consistent with RG 1.177 (Reference 2) guidance, for the CRMP model:

- A process for evaluation and disposition of proposed facility changes shall be established for items impacting the CRMP model (Section 2.3.4, Item 7.2).
- The CRMP model shall accurately reflect the as-built, as-operated plant consistent with RG 1.200 guidance for PRA capability category II (Section 2.3.5, Item 9 and Section 4.1).
- The CRMP model shall be updated to reflect the as-built and as-operated plant on a periodic basis not to exceed two refueling cycles (Section 2.3.5, Item 9.1).

Common cause treatment, as applied in the CRMP model, shall be consistent with the PRA model and Risk Managed Technical Specification (RMTS) guidance. If a component is out-of-service for planned maintenance, there is no justification for changing the common cause failure (CCF) factors. If an emergent failure occurs, the "extent of condition" evaluation performed by Operations either addresses the situation or provides assurance that a CCF is not occurring, so no changes in CCF modeling are necessary. However, optionally if an “extent of condition” evaluation cannot establish with a high degree of confidence that there is no common cause failure mechanism, the probability that the redundant component is failed from a common cause failure mechanism will be modified numerically, consistent with the guidance in RG 1.177 while calculating the RICT. If, for either option, it is determined that a common cause failure mechanism exists, the redundant SSC will be declared inoperable and cannot be considered available for PRA functional. The previously mentioned set of procedures/instructions ensures that basic events for CCF of multiple components will not be changed within the CRMP model by excluding (removing) them from the “tag table” (Section 2.3.4, Item 6). Specifically, the treatment of CCF in the CRM Tool will be as described below:

- Planned Configurations:
  - For planned configurations the RICT calculations will be performed consistent with NEI 06-09, Section 3.3.6, “Common Cause Failure Consideration,” guidance on the treatment of CCF, as follows:

    “For all RICT assessments of planned configurations, the treatment of common cause failures in the quantitative CRM Tools may be performed by considering only the removal of the planned equipment and not adjusting common cause failure terms.”

- This approach will result in slightly shorter completion times than if RICTs were calculated using the RG 1.77 approach (i.e., it is conservative), and it will prevent deviation from the NRC's approach of NEI 06-09.

o Emergent Configurations

- For emergent configurations, the RICT program will abide by NEI 06-09, Section 3.3.6, "Common cause Failure Consideration" guidance on the treatment of CCF, as follows:

  "For RICT assessments involving unplanned or emergent conditions, the potential for common cause failure is considered during the operability determination process. This assessment is more accurately described as an 'extent of condition' assessment."

  "In addition to a determination of operability on the affected component, the operator should make a judgement with regard to whether the operability of similar or redundant components might be affected."

  "The components are considered functional in the PRA unless the operability evaluations determines otherwise."

- An "extent of condition" evaluation together with an operability evaluation will provide an assessment of the vulnerability of the operable redundant components to any common cause failure potential. The RICT determination process for an emergent configuration will be consistent with the following guidance provided in the NRC SER for NEI 06-09:

  *"**Emergent Failures.** During the time when a RICT is in effect and risk is being assessed and managed, it is possible that emergent failures of SSCs may occur, and these must be assessed to determine the impact on the RICT. If a failed component is one of two or more redundant components in separate trains of a system, then there is potential for a common cause failure mechanism. Licensees must continue to assess the remaining redundant components to determine there is reasonable assurance of their continued operability, and this is not changed by implementation of the RMTS. If a licensee concludes that the redundant components remain operable, then these components are functional for purposes of the RICT. However, the licensee is required to consider and implement additional risk management actions (RMAs), due to the potential for increased risks from common cause failure of similar equipment. The staff interprets TR NEI 06-09, Revision 0, as requiring consideration of such RMAs whenever the redundant components are considered to remain operable, but the licensee has not completed the extent of condition evaluations…"*

- In keeping with the above NRC guidance, if it is determined that redundant components remain operable, these components are considered PRA functional for purpose of RICT determinations. However, FNP will consider and implement additional RMAs, due to the potential for increased risks from common cause failure of similar equipment, whenever the redundant components are considered to remain operable but an extent of condition evaluation has not yet been completed. The consideration and implementation of additional RMAs, according to the NRC SER on

NEI 06-09, is considered to be consistent with the guidance of RG 1.177 regarding the treatment of increased risks from common cause failures.

"TS Loss of Function Conditions (LOF)**"** A RICT is allowed to be calculated during a TS LOF Condition if at least one train in a two train system is PRA functional (for more than two train systems, the number of trains that are required to be PRA functionality is described in Enclosure 1, Table E1-1). However, the following additional constraints shall be applied to the criteria for "PRA Functional".

1. Any SSCs credited in the PRA Functionality determination shall be the same SSCs relied upon to perform the specified Technical Specifications safety function unless such SSCs have been approved by the NRC for performance of TSs safety function.
2. Design basis success criteria parameters shall be met for all design basis accident scenarios for establishing PRA Functionality during a Technical Specifications loss of function condition where a RICT is applied.
3. The RICT for these loss of function conditions may not exceed 24 hours.
4. If a TS LOF is due to CCF vulnerability of the redundant train(s) and does not impact the PRA functionality of the redundant train(s), a RICT can only be established if the inoperability of the initial TS Condition is considered PRA Functional.

- Criteria shall exist to require CRMP model updates concurrent with implementation of facility changes that significantly impact RICT calculations (Section 2.3.5, Item 9.2).

- Initiating event models in the CRMP shall accurately include external conditions and effects of out-of-service equipment (Section 2.3.5, Item 1).

- The impacts of out-of-service equipment shall be properly reflected in the CRMP model initiating event models, as well as system response models. For example, if a certain component being declared inoperable and placed in a maintenance status is modeled in the PRA, the entry of that equipment status into the CRMP model must accommodate risk quantification to include both initiating event and system response impact (Section 4.2, Item 1).

- The CRMP model fault trees shall be traceable to the PRA (Section 2.3.5, Item 3).

- Changes to the CRMP model and data shall correctly reflect configuration-specific risk (Section 4.2, Item 3).

- In order for human recovery actions as modeled in the PRA to be credited in the RICT Program, such actions shall be performed via approved station procedures with the implementing personnel trained in their performance (Section 4.2, Item 4).

- The baseline PRA models assess average annual risk. However, some risk is not consistent throughout the year, and the CRMP tool needs to properly assess change in risk for the existing plant configuration. The departmental procedure process requires that time averaging features of the baseline PRA shall be excluded from the CRMP model (specific items discussed in Table E6.2) (Section 2.3.4, Item 5).

- Benchmarking of the CRMP model against the baseline PRA is performed and documented to demonstrate consistency (Section 2.3.5., Item 3).

4.0 Quality Requirements

Southern Nuclear Operating Company (SNC) employs a multi-faceted approach to establishing and maintaining the quality of the PRA models, including the CRMP models, for all operating SNC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process (described in Enclosure 5) and the use of self-assessments and independent peer reviews (described in Enclosure 2). The information provided in Enclosure 2 demonstrates that the FNP at-power internal events PRA model (including internal flooding) and internal fire PRA are consistent with RG 1.200, Revision 2 (Reference 3), requirements. This information provides a robust basis for concluding that the PRA is 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, and changes made to the CRMP configuration files, are controlled and documented by departmental procedures. Those procedures specify an acceptance test to be performed after every CRMP model update. This testing verifies 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 included in CRMP to the correct basic events in the CRMP model.

Prior to the implementation of the RICT Program, results of the acceptance testing for the integrated single top model, including fire (Step 6), the optimized single top model if developed (Step 7), and the CRMP model, which is used for configuration specific calculations (Step 8) are compared with the model produced in Step 5 to ascertain fidelity of the CRMP model. The results are documented in the model development reports and/or calculations.

The model development reports will discuss the results and justify variations in the CDF and LERF results. The primary variations in the CDF and LERF results typically arise from using average maintenance models in Step 5 and zero-maintenance models in Step 8.

5.0 Training and Qualification

The training for personnel developing the CRMP model used to support RICT Program implementation is developed based on SNC procedures as described in Enclosure 8. The qualification of personnel developing and using the CRMP model is controlled by SNC qualification and training programs based on the Institute of Nuclear Power Operation (INPO) Accreditation (ACAD) requirements. SNC fleet-wide procedures establish the responsibilities and requirements for the training and qualification of personnel who perform engineering activities. The following discussion provides an overview of general accountabilities and aspects of FNP training programs applicable to plant staff involved with the CRMP tool development and use.

The Southern Nuclear Fleet Operations Training Manager is accountable for the performance and use of Training procedures. Site Functional Area Managers are responsible for the following:

- Governance and oversight of any site-specific sub-tiered instructions, guidelines, and forms and the overall administration of and performance of the continuing training program
- Conducting courses to support the training and qualification of individuals in the engineering population.
- Ensuring that training and qualification records are processed in accordance with procedures.

The SNC Training Manager is responsible for conducting courses to support the training and qualification of individuals in the Engineering population and for processing Training and Qualification records in accordance with SNC fleet-wide procedures and applicable site procedures.

The Engineering Fleet Training Program Committee (TPC) is composed of the four Engineering TPC Chairs, one Training Manager, and the Vice President of Engineering. This group is responsible individually and collectively to drive training program performance to levels of excellence and leverage training to drive station performance to levels of excellence. They are responsible for ensuring:

- Training program performance issues are identified and resolved
- Student performance shortfalls (in training and in-plant) are identified and resolved
- Training is a core business and addresses needs through annual, biennial and long-range planning.
- Overall training program health remains strong and meet station needs, and provides workers the knowledge and skills necessary for job performance.
- Approving position-specific qualifications that are designated for common fleet Engineering duties and activities.

FNP Site and Corporate Department Managers with personnel performing Job Performance Requirements (JPRs) that are covered by the Training program are responsible for the following:

- Ensuring that individuals are evaluated for inclusion in, or exclusion from, the Engineering Training program population based on their job assignment.
- Ensuring that personnel in their department complete qualifications and training in accordance with procedural requirements.
- Maintaining and reviewing the qualification requirements in the Learning Management System (LMS).
- Administering the Engineering Training Population Determination Form for Supervisors who perform engineering activities independently or who perform the Final Technical Review of engineering activities. This applies regardless of inclusion in or exclusion from the Engineer population.
- Ensuring that only qualified individuals perform engineering activities independently or perform the Final Technical Review of engineering activities. This applies to individuals regardless of inclusion in or exclusion from the Engineer population.
- Designating one or more individuals as Department Training Coordinator(s) to ensure effective use of LMS.
- Designating personnel to be Technical Mentors.
- Participating in Engineering Training Committees, which oversee the Engineering Support Personnel Accredited Training Program.
- Coordinating the scheduling of assignments

Each Supervisor with personnel performing JPRs covered by the Training program is responsible for the following:

- Checking employee qualifications prior to assigning work, to ensure that the assigned personnel are qualified for the work being assigned
- Ensuring items and qualifications are assigned, as needed, to assigned personnel
- Participating in Engineering Training Committees, which oversee the Engineering Support Personnel Accredited Training Program.

Personnel performing Engineering JPRs that are covered by the training program are responsible for the following:

- Verifying they are qualified in the Learning Management System (LMS) prior to independently performing the work, whether or not they are in the accredited program population.

- Ensuring that completion of qualifications and training is done in accordance with procedural requirements.

As stated above, the qualification of personnel developing and using the CRMP model is controlled by the SNC qualification and training programs, which are based on INPO ACAD requirements.

6.0 References

1. NEI 06-09, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,” Nuclear Energy Institute, Revision 0-A, October 2012 (ADAMS Accession No. ML 122860402).

2. Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” August 1998 (Adams Accession No. ML003740176).

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.

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines"

Enclosure 7

Key Assumptions and Sources of Uncertainty

Table of Contents

1.0 Introduction

The purpose of this enclosure is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for the Risk Informed Completion Time (RICT) Program. Topical Report NEI 06-09 (Reference 1), Section 2.3.4, item 10 requires an evaluation to determine insights that will be used to develop risk management actions (RMAs) to address these uncertainties. The baseline internal events (including internal flooding) PRA and fire PRA (FPRA) models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to the RICT Program calculations, to perform sensitivity analyses where appropriate, to discuss the results and to provide dispositions for the RICT Program.

The epistemic uncertainty analysis approach described below applies to the internal events PRA, and any epistemic uncertainty impacts that are unique to FPRA are also addressed. In addition, Topical Report NEI 06-09 requires that the uncertainty be addressed in RICT Program Configuration Risk Management Program (CRMP) tools by consideration of the translation from the PRA model to the CRMP tool. The CRMP model, is discussed in Enclosure 6. It consists of separate zero-maintenance annual average PRA models which include the internal events (including internal flooding) PRA model, internal fire events PRA model that reflects NFPA 805 plant modifications, seismic bounding delta CDF/LERF values, and also main control room abandonment bounding delta CDF/LERF values that are calculated separately from the fire PRA model when a TS inoperable SSC needed for remote shutdown, consistent with plant operating procedures, is determined not to be PRA functional. The main control room abandonment bounding delta CDF/LERF option will no longer be used after its contribution is fully integrated into the fire PRA model. The CRMP model that reflects the as-built and as-operated plant condition including credit for NFPA 805 modifications will be developed prior to implementation of the RICT Program. The model translation uncertainties evaluation and impact assessment are limited to new uncertainties that could be introduced by application of the CRMP tool during RICT Program calculations.

2.0 Assessment of Internal Events PRA Epistemic Uncertainty Impacts

In order to identify key sources of uncertainty for RICT Program application, the internal events baseline PRA model uncertainty report was developed, based on the guidance in NUREG-1855 (Reference 2). As described in NUREG-1855, sources of uncertainty include “parametric” uncertainties, “modeling” uncertainties, and “completeness” (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the Farley Nuclear Plant (FNP) baseline PRA model aleatory uncertainty analysis as part of the baseline model development and quantification (Reference 3).

Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. The assumptions are defined consistent with the definition provided in NUREG-1855. Plant-specific assumptions made for each of the FNP internal events PRA technical elements are collected from each portion of the PRA model development and quantification and evaluated for the base PRA.

The evaluation considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. The Electric Power Research Institute (EPRI) compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (EPRI 1016737, Reference 5) and an evaluation of each generic source of modeling uncertainty was performed (Reference 4).

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness have been identified relative to the TSTF-505 application, based on the results of the internal events PRA and fire PRA peer reviews.

Based on following the methodology in Reference 5 for a review of sources of uncertainty, the potential sources of uncertainty and impact of these items on RICT program implementation are discussed in Table E7.1 relative to the need for consideration as key sources of uncertainty for the RICT application that might warrant treatment through additional RMAs. Note that RMAs will be developed when appropriate using insights from the PRA model results specific to the configuration.

Based on the evaluation summarized in Table E7.1, none of the evaluated sources represents a key source of uncertainty for the RICT application.

Although not addressed in Table E7.1 through review of the base model, the RICT process addresses possible uncertainty in the reliability of SSCs considered to be PRA Functional. In cases where SSC degradation may be the cause of inoperabilities, PRA Functionality determinations are performed consistent with the following NEI 06-09 guidance:

“The PRA function may be considered in cases that involve SSC inoperabilities which, while degraded, do not involve a potential for further degrading component performance. In most cases,

degrading SSCs may not be considered to be PRA functional while inoperable. For example, a pump which fails its surveillance test for required discharge pressure is declared inoperable. It cannot be considered functional for calculation of a RICT, since the cause of the degradation may be unknown, further degradation may occur, and since the safety margin established by the pump's operability requirements may no longer be met. As a counter example, a valve with a degrading stroke time may be considered PRA functional if the stroke time is not relevant to the performance of the safety function of the valve; for example, if the valve is required to close and is secured in the closed position."

As a result, the failure probability need not be increased depending on the failure mechanism causing the degraded condition. The SSC's nominal reliability remains applicable and consistent with the definition of PRA functionality in NEI 06-09 0-A, process requirement number 11.1.2 (i.e., further degradation that could impact PRA functionality is not expected during the RICT). Given an inoperable condition caused by a degraded condition, the FNP RICT Program allows only two choices to be made in the CRM Tool:

- Either a "PRA non-functional" or "PRA functional" condition to represent the TS degraded condition.
    - If the inoperability is evaluated as a "PRA non-functional" condition, CRM Tool will treat the SSC as failed, or
    - If the inoperability is evaluated as a "PRA functional" condition, CRM Tool will treat the SSC with the nominal base-case failure probability.

The rationale for using the nominal reliability for a PRA functional SSC includes the determination that the base case PRA results are still applicable, the degraded condition has been demonstrated to meet the PRA success criteria, and the SSC is considered fully available. No additional uncertainty or sensitivity analysis is planned to be performed during a RICT entry, which is consistent with the expectations of NEI 06-09 and the NRC SER on NEI 06-09.

The baseline PRA does not include seasonal variations from hazards but there are certain initiating events that can be affected by seasonal variations (e.g., loss of offsite power). The assumptions involve applying the generic industry frequency for the loss of offsite power event developed in NUREG/CR-6890 (Reference 14). The RICT Program will include a qualitative consideration of weather events as part of the RMA decision process when LCO 3.8.1 CTs are extended to address this source of uncertainty.

3.0 Assessment of Translation Uncertainty Impacts

Modification of the baseline PRA models is required to create the CRMP model used for RICT Program calculations. These modifications, described in Enclosure 6, may introduce new sources of model translation uncertainty. Table E7.2 provides a description of the model changes and dispositions of whether any of the model changes made represent possible new sources of model uncertainty that must be addressed.

Table E7.1

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Loss Of Offsite Power (LOSP) frequency and fail to recover offsite power probabilities	LCOs for which LOSP scenarios have an effect on the RICT	The LOSP frequency and fail to recover offsite power probabilities are based on available industry data. The overall approach for the LOSP frequency and fail to recover probabilities utilized is consistent with industry practice. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations. However, the RICT Program will include a qualitative consideration of weather events as part of the RMA decision process when LCO 3.8.1 CTs are extended to address this source of uncertainty.
Reactor coolant pump (RCP) seal LOCA modeling	Potentially all LCOs in the RICT program	The plant has been modified to install the RCP shutdown seals developed by Westinghouse to reduce the likelihood of RCP seal leakage beyond normal values. The shutdown seal is modeled consistent with WCAP-17100 (Reference 8). Consequential RCP Seal failure as a result of loss of seal cooling is treated through the fault tree structure. Because a consensus industry seal LOCA model endorsed by the NRC is used, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table E7.1

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Failure of core cooling following containment failure is not explicitly modeled	LCOs for which loss of containment heat removal scenarios have an effect on the RICT	A combination of generic and plant-specific analyses are used to evaluate the impact of containment failure on ECCS recirculation. Failure of ECCS recirculation as a consequence of containment overpressure or isolation failure is not modeled. Since the Farley design basis does not credit containment overpressure in the RHR pump NPSH analysis, and the Farley PRA requires operable cooling through the RHR heat exchangers or containment fan coolers for success of ECCS recirculation, the loss of NPSH due to steam release from an unisolated containment is considered unlikely. Therefore, this is not a key source of uncertainty and will not be an issue for RICT calculations.

Table E7.1

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
The diesel generator switchgear room coolers are not included in the PRA model	LCOs for which the availability of on-site ac power have an effect on the RICT	Room heat up calculations performed for the diesel generator switchgear rooms shows that the realistic equipment operability temperature limit appropriate for the switchgear rooms was not exceeded after 24 hour loss of ventilation (Reference 16) This may represent a source of model uncertainty because the TS equipment operability limit is lower than the realistic operability limit. The contribution of room cooling failures to DG switchgear failure will not be an issue for delta risk applications, i.e., including the room coolers in the model would affect the baseline and RICT configuration in the same manner and not significantly impact the delta risk calculations. Further, the modeling is conservative for RICT in the sense that if the DG or DG switchgear were declared inoperable due to room cooling issues, a PRA functionality determination could not be made without appropriately including the room coolers and associated support equipment, and any necessary operator actions into the model. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table E7.1 Assessment of Internal Events PRA Epistemic Uncertainty Impacts		
Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Credit for battery life out to two hours based on conservative FSAR analysis without explicit representation of or credit for successful load shedding	LCOs for which the availability of on-site ac power have an effect on the RICT	The two hour battery life assumes procedurally-directed load shedding has not been implemented. Without recovery of DC power at two hours, equipment requiring DC power (e.g., turbine-driven AFW pump (TDAFW)) is assumed unavailable after battery depletion. However, realistically assessing battery life involves other uncertainties and is complex. Although this may represent a source of model uncertainty, it is unlikely to be an issue for delta risk applications, since the DC supply to the TDAFW pump has a four-hour rating and manual action could be taken to maintain the steam admission valves open beyond 2 hours (Reference 15). Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table E7.1

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
The use of a single value in the PRA model for unrecoverable failure due to sump screen plugging for all sequences.	Potentially all LCOs in the RICT program	There is not a consistent method for the treatment of ECCS sump performance. Unrecoverable failure of recirculation due to sump screen plugging is included in the model for each sump intake based on NUREG/CR- 4550 (Reference 9). Although this may represent a moderate source of model uncertainty, it is not an issue for delta risk applications since sump screen plugging is not TS-specific so that the assumption affects both the baseline and RICT calculations equally. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.
Assumption that failure of pressure relief (if required) is negligible and can be ignored in the success criteria for all sequences except ATWS.	Potentially all LCOs in the RICT program	For transients other than ATWS, there is significant redundancy in RCS pressure relief capability, such that the likelihood of pressure relief failure is small and is unlikely to be an issue for delta risk applications. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table E7.1

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Treatment of thermally-induced SGTR, and no credit for RCS depressurization by induced hot leg or surge line failure and subsequent in-vessel injection.	LCOs for which the LERF results may have some effect on the RICT	During high-pressure core damage scenarios, a "race" occurs to determine where the RCS will first fail. While the reactor vessel will eventually fail as the molten core degrades the lower vessel head, failures may also occur in the steam generator tubes (discussed below) or in the hot leg or surge line of the reactor coolant system. For high- pressure, station-blackout-like scenarios which tend to occur on this branch, the likelihood of hot leg failure is very high. Pressure induced and thermally induced SGTR are modeled as separate events in the Level 2 event tree. If an induced SGTR does not occur then hot leg/surge line failure is evaluated in the event tree. The approach used is consistent with industry practice, in accordance with WCAP-16341-P (Reference 10). Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table E7.1

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Interfacing Systems LOCA (ISLOCA) frequencies	Potentially all LCOs in the RICT program	A detailed ISLOCA analyses was performed that involved screening of potential ISLOCA pathways, calculation of the frequency of failure of the high pressure/low pressure interface of each unscreened interfacing system, and calculation of the probability of piping or component failure in the interfacing system as a result of the exposure to high pressure. Calculations were performed to assess the failure frequency of each scenario based on its specific configuration. These calculations are based on NSAC-154 (Reference 11) and NUREG/CR-5102 (Reference 12) with modifications as appropriate to represent differences in the Farley configuration. The impacts of overpressure on each of the above ISLOCA scenario pathways were evaluated using the guidelines of NUREG/CR-5862 (Reference 13). The approach for the ISLOCA frequency determination applies state-of-the art methods. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table E7.1

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Treatment of flow diversion paths	Potentially all LCOs in the RICT program	In the PRA model, diverted flow paths in fluid systems are removed if the cross-sectional area of the diversion path is less than ten percent of the cross-sectional area of the main process flow path, and potential flow diversion paths that are greater than one third (1/3) the diameter of the main flow path should be further evaluated. This approach does not explicitly treat pressure effects of flow diversions from high pressure to low pressure, and no supporting thermal hydraulic analyses are performed to assess the validity of this assumption for these cases. This should not be an important source of model uncertainty in most applications, particularly delta-risk applications, since the flow diversion assumptions are not TS-specific, and the assumption affects both the baseline and RICT calculations equally. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table E7.1

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Epistemic Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Human Error Probabilities (HEPs): Uncertainties associated with the assumptions and method of calculation of HEPs for the Human Reliability Analysis (HRA) may introduce uncertainty. Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	Potentially all LCOs in the RICT program	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations. Refer to Enclosure 10 for additional discussion on risk management actions (RMAs).

Table E7.2
Assessment of Translation Uncertainty Impacts

EOOS or Similar CRMP Model	Part of Model Affected	Impact on Model	Disposition
Model logic structure optimized if required to increase solution speed. <u>Analysis Assumptions</u>: None	Event trees, one-top model structure, inserted fire initiating events	The restructured model is logically equivalent and produces results comparable to the baseline PRA logic model	Since the restructured model produces comparable numerical results, this is not a source of uncertainty.
Incorporation of seismic bias to support RICT Program risk assessment calculations as FNP does not include a seismic PRA. <u>Analysis Assumptions</u>: A conservative value for seismic delta CDF is applicable.	Calculation of RICT and RMAT within EOOS or similar CRMP model	The addition of a bounding impact for seismic events has no impact on baseline PRA or CRMP model since it is added as an additional delta risk contribution. Impact is reflected in calculation of RICT and RMAT.	Since this is a bounding approach for addressing seismic risk in the RICT Program, it is not a source of uncertainty, and RICT Program calculations are not impacted, so no mandatory RMAs are required. The use of bounding approach is acceptable per NEI 06-09 guidance.
Set plant availability (PAV) basic event to 1.0. <u>Analysis Assumptions</u>: None	Basic event PAV	Since the CRMP model evaluates specific configurations during at-power conditions, the use of a PAV factor less than 1.0 is not appropriate. This change allows the CRMP model to produce accurate results for specific at-power configuration.	This change is consistent with CRMP tool practice; therefore this change does not represent a source of uncertainty, and RICT Program calculations are not impacted, so no mandatory RMAs are required.

4.0 Assessment of Supplementary Fire PRA (FPRA) Epistemic Uncertainty Impacts

The purpose of the following discussion is to address the epistemic uncertainty in the FNP FPRA (Reference 6). The FNP FPRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the FPRA and because the state of knowledge in these elements continues to evolve. The Farley FPRA was developed using consensus methods outlined in NUREG/CR-6850 (Reference 7) and interpretations of technical approaches as required by NRC for approval of the NFPA-805 application. Enclosure 2 provides a detailed discussion of the Peer Review F&Os and the resolutions.

FNP used guidance provided in NUREG-1855 (Reference 2) to address uncertainties associated with FPRA for the RICT Program application. As stated in Section 1.5 of NUREG-1855:

> "Although the guidance does not currently address all sources of uncertainty, the guidance provided on the process for their identification and characterization and for how to factor the results into the decision making is generic and is independent of the specific source. Consequently, the process is applicable for other sources such as internal fire, external events, and low power and shutdown."

NUREG-1855 also describes an approach for addressing sources of model uncertainty and related assumptions. It defines:

> "A source of model uncertainty is one that is related to an issue in which no consensus approach or model exists and where the choice of approach or model is known to have an effect on the PRA (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion and introduction of a new initiating event)."

NUREG-1855 defines consensus model as:

> "A model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that NRC has utilized or accepted for the specific risk-informed application for which it is proposed."

The potential sources of model uncertainty in the FPRA model were characterized for the 16 tasks identified by NUREG/CR-6850. This framework was used to organize the assessment of baseline FPRA epistemic uncertainty and evaluate the impact of this uncertainty on RICT Program calculations. Table E7.3 outlines sources of uncertainties by task and their disposition.

The results of this assessment concluded that no sensitivity analyses were needed.

Table E7.3
Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

Task #	Description	Sources of Uncertainty	Disposition
1	Analysis boundary and partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	The methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.
2	Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized though use of the PWROG Generic MSO list and the process used to identify and assess potential MSOs. The methodology for the Component Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.
3	Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. Some systems are not credited in the FPRA and are therefore treated as being failed everywhere. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	The methodology for the Cable Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.

Table E7.3
Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

Task #	Description	Sources of Uncertainty	Disposition
4	Qualitative Screening	Qualitative screening was not performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables) identified in the prior two tasks and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.
5	Fire-Induced Risk Model	A reactor trip is assumed as the initiating event for all quantification. The FPRA does not consider any special initiators (like loss of Service Water or Instrument Air) and does not consider turbine trip/MSIV closure events even though they may occur in a limited number of fire scenarios.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process has reviewed all significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.

Table E7.3
Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

Task #	Description	Sources of Uncertainty	Disposition
6	Fire Ignition Frequency	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates.	Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.
7	Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The Farley FPRA development did not screen out any fire initiating events based on low CDF/LERF contribution. The methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.
8	Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are 8 and 11. The discussion of uncertainty for both tasks is provided in the discussion for Task 11.	See Task 11 discussion.

Table E7.3 Assessment of Supplementary FPRA Epistemic Uncertainty Impacts			
Task #	Description	Sources of Uncertainty	Disposition
9	Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. The methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.

Table E7.3
Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

Task #	Description	Sources of Uncertainty	Disposition
10	Circuit Failure Mode Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability is assigned using industry guidance such as that published in NUREG/CR-6850. The uncertainty associated with the applied conditional failure probabilities poses competing considerations. On the one hand, a failure probability for spurious operation could be applied based solely on cable scope without consideration of less direct fire affects (e.g., a 0.3 failure likelihood applied to the spurious operation of a motor-operated valve (MOV) without consideration of the fire- induced generation of spurious signal to close or open the MOV). The analysis has biased the treatment such that it is assumed the spurious signal will always drive the valve in the unsafe direction. In addition, for those valves that might have multiple desired functions – consideration of spurious closure and consideration of failure to open on demand, the non-spurious failure state is treated with a logical TRUE rather than the complement of the spurious probability. For those valves that only have an active function, the potential for a spurious signal to drive the valve in the desired direction is ignored. The treatment results in skewing of the results such that the resulting risk is over-estimated.	Uncertainty in the circuit failure mode likelihood analysis could lead to assumed failures of related components and related system functions. This would generate conservative results and that would typically be acceptable for most applications. Furthermore, a consensus modeling approach is used for Circuit Failure Mode Likelihood Analysis. Circuit failure mode likelihood analysis was generally limited to those components where spurious operation could not be caused by the generation of a spurious signal. This approach limited the introduction of non-conservative uncertainties. For the ‘simple’ cases, the potential exists for assuming a failure likelihood greater than 0 in some areas where the cables capable of causing spurious operation are not located. Additional refinement to this approach was performed, as necessary, on risk significant scenarios. So the application of circuit failure probabilities is considered to have minimal impact on the results. The methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.

Table E7.3
Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

Task #	Description	Sources of Uncertainty	Disposition
11	Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and the response of plant staff (detection, fire control, and fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	Consensus modeling approach is used for the Detailed Fire Modeling. Detailed fire modeling was performed only on those scenarios which otherwise would have been notable risk contributors and only where removal of conservatism in the generic fire modeling solution was likely to provide benefit either via a smaller zone of influence or to credit automatic suppression. Fire modeling was used to evaluate the time to abandonment for control room fire scenarios for a range of fire heat release rates. The analysis methodology conservatism is primarily associated with conservatism in the heat release rates specified in NUREG/CR-6850. A review of the generic fire modeling treatment summary zone of influence data indicates that the reduction in zone of influence is possible for smaller fires, through additional refinement of fire scenarios can be pursued using multi-point analysis of the heat release rates as opposed to the use of a bounding fire, is not significant. The potential for this slightly reduced zone of influence to reduce the consequences associated with the smaller fire is very small. Without a reduction in consequences a multi-point treatment of the heat release rate curves would have no impact on results. The methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.

Table E7.3
Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

Task #	Description	Sources of Uncertainty	Disposition
12	Post-Fire Human Reliability Analysis	There are relatively few HFEs of high importance in the FPRA model. Conservative human error probability (HEP) adjustments were made to the nominal HEP values used in the FPIE model then revisited to address unique fire considerations. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The human error probabilities were calculated using the EPRI HRAC and included the consideration of loss of necessary cues due to fire. The impact of any remaining uncertainties is expected to be small. The methodology for the Post-Fire Human Reliability Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.
13	Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified FPRA model. The methodology for the Seismic- Fire Interactions Assessment task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.

Table E7.3
Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

Task #	Description	Sources of Uncertainty	Disposition
14	Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit. However, the selected truncation limit is several orders of magnitude below the typical CDF value calculated, and is consistent with the requirements of the PRA Standard.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. The methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.
15	Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. The methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.
16	FPRA Documentation	This task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements. The methodology for the FPRA documentation task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no mandatory RMAs are required.

As noted above, the FNP FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC for approval of the NFPA-805 application. Therefore, consistent with NUREG-1855 guidance, FPRA modeling does not introduce any epistemic uncertainties that would require sensitivity treatment to the RICT Program risk assessment calculations.

5.0 References

1. Topical Report NEI 06-09, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document,” Revision 0-A, November 2006.

2. NUREG-1855, “Guidance on the treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” Main Report, March 2009.

3. Farley Nuclear Plant Units 1 and 2 Internal Events PRA Model: Revision 9 Version 3, PRA-BC-F-14-001, January 9, 2014

4. Appendix D to F-RIE-IEIF-U00-013, “Characterizing the Sources of Model Uncertainty for Farley Nuclear Plant,” September 2011.

5. TR-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments,” Electric Power Research Institute, Final Report, December 2008.

6. Farley Nuclear Plant Unit 1 Fire Probabilistic Risk Assessment Summary Report: Revision 5, PRA-BC-F-11-017, August 2014.

7. EPRI/NRC-RES, Fire PRA Methodology for Nuclear Power Facilities, EPRI1011089-NUREG/CR-6850, August 2005.

8. WCAP-1700-NP, “PRA Model for the Westinghouse Shut Down Seal,” Revision 1, Non-proprietary, Westinghouse Electric Company, January 2010.

9. NUREG/CR-4550, “Analysis of Core Damage Frequency From Internal Events: Methodology Guidelines,” Revision 1, Sandia National Laboratory for U.S. NRC, 1989

10. WCAP-16341-P, Revision 0, “Simplified Level 2 Modeling Guidelines,” Westinghouse Electric Company, November 2005.

11. NSAC-154, “ISLOCA Evaluation Guidelines,” ERIN Engineering and Research, Inc. (prepared for Electric Power Research Institute), September 1991.

12. NUREG/CR-5102, “Interfacing Systems LOCA: Pressurized Water Reactors,” Brookhaven National Laboratory for U.S. NRC, February 1989.

13. NUREG/CR-5862, “Screening Methods for Developing Internal Pressure Capacities for Components in Systems Interfacing with Nuclear Power Plant Reactor Coolant Systems,” U.S. NRC, May 1992.

14. NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," U.S. NRC, December 2005

15. Joseph M. Farley Nuclear Plant Units 1 and 2, “Accident Sequence Notebook PRA Model Revision 9,” Version 1.2, September 2010.

16. NAI-1551-006, Revision 1, Farley Nuclear Plant Units 1 and 2 Auxiliary and Control Building GOTHIC Room Heat-up Analysis PRA cases, August 2014.

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines"

Enclosure 8

Program Implementation

Table of Contents

1.0 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 the personnel required for implementation of the RICT Program. Several procedures and processes are detailed in Enclosures 5, 9, and 10; those discussions are not repeated as part of this enclosure. Those topics include Probabilistic Risk Assessment (PRA) Maintenance and Update process (Enclosure 5), Cumulative Risk Assessment and Performance Monitoring Program (Enclosure 9), and Risk Management Actions (Enclosure 10).

2.0 RICT Program Procedures

The procedures discussed below were developed for implementing the RICT Program for the SNC fleet, and are currently in effect for Vogtle Electric Generating Plant. They will be adopted for use in implementing the FNP RICT program. They provide guidance to the appropriate SNC personnel on the following topics:

- On-Line Configuration Risk Management Program (CRMP, Reference 2):

  This procedure provides requirements for Implementation of the RICT program while in Modes 1 & 2. In addition, it provides requirements for outlining planning and scheduling strategies to minimize risk (in terms of core damage frequency (CDF, ICDP) and large early release frequency (LERF,ILERP)), and meeting requirements necessary for maintaining and retaining a chronological history of configuration changes and their risk impacts (in terms of CDF,ICDP,LERF, and ILERP) throughout the operating cycle

- Risk Management Actions (RMAs) for the RICT Program (Reference 3):

  This instruction provides requirements for development and implementation of RMAs for the RICT program.

- Calculation of RMAT and RICT for the RICT Program (Reference 4):

  This procedure provides detailed requirements and limitations of the RICT Program at Southern Nuclear Company. It includes the calculation of RICT and RISK MANAGEMENT ACTION TIMES (RMAT). This procedure is applicable to sites that have an approved license amendment to use the RICT Program.

- PRA Functionality Determination (Reference 5):

  This procedure provides requirements for determining whether structures, systems and components (SSCs) that are declared inoperable can be considered PRA FUNCTIONAL in RICT calculations.

- Recording Limiting Conditions for Operation (Reference 6):

  This procedure provides instructions to the control room operator for using the interface between the control room electronic narrative log and CRMP.

The procedures discussed above may be revised or supplemented by other procedures, as deemed necessary to implement the RICT Program effectively at FNP Units 1 and 2. They are described in more detail below.

2.1 On-Line Configuration Risk Management Program

This procedure (Reference 2) describes, in general terms, the CRMP, as it pertains to the RICT Program as well as parts of the 10 CFR 50.65(a)(4) program. It is the parent procedure for both these programs.

With respect to the RICT Program, this procedure has the following attributes:

- Identifies the plant management individual with the authority to approve entry into the RICT Program.
- Details the plant conditions under which the RICT Program is applicable.
- Acts as the overarching guidance for the SNC risk assessment and risk management procedures.
- Contains important definitions for the RICT Program.
- Details many of the requirements, per NEI 06-09 Revision 0-A (Reference 1), for the RICT Program.
- Identifies departmental and position responsibilities within the RICT program.
- Outlines the requirement to identify and implement Risk Management Actions (RMAs) when the RMAT is exceeded or is anticipated to be exceeded.
- Describes the necessary attributes for the SNC CRMP tool.

The above guidance is consistent with NEI-06-09 (Reference 1).

The CRMP procedure is maintained as a SNC procedure. It is managed by the Fleet Work Control Manager and is under the ownership of Fleet Work Management (FWM). The ownership of this procedure is subject to change if deemed appropriate. This procedure is currently designated as applicable only to Vogtle Units 1 and 2. Upon approval of the RICT program for FNP, the procedure will be revised to note that it is also applicable to FNP Units 1 and 2.

2.2 Risk Management Actions for the Risk Informed Completion Time Program

This procedure (Reference 3) describes the risk assessment and management processes for the SNC fleet of nuclear plants. It provides general guidelines for the risk assessment and management of maintenance activities, both planned and emergent. This procedure is a sub-tier procedure to the On-Line CRMP, described above.

Risk Management Actions are targeted toward RMA candidates in order to manage and control increases in CDF/LERF attributed to internal events, fire events, and other hazards modeled in CRMP which include the following:

- Identify RMA candidates which identify SSCs, initiating events and fire zone considered important for a given plant configuration when a RICT is implemented.
- Develop RMAs using RMA candidates and develop additional RMAs, as appropriate.
- Communicate RMAs to Operations, Fire Protection personnel, and other affected departments to facilitate RMA planning and RMA implementation.
- Implement RMAs for conditions which require RMA implementation as required prior reaching RMAT.
- Document implementation of RMAs in the Control Room Narrative Log. The time of actual implementation and removal of RMAs should be documented. This may be accomplished in multiple log entries.

The risk management procedure also indicates that while the Outage and Scheduling department is responsible for the planning, scheduling and assessing planned maintenance items, the site Operations department is primarily responsible for the evaluation of emergent work for the RICT Program. This evaluation includes the risk assessments and the calculation of the RMATs and RICTs.

The Risk Management Actions program guidance is maintained as a SNC procedure. It is primarily utilized by the Outage and Scheduling and Operations Departments under the ownership of Fleet Work Management. The ownership of this procedure is subject to change if deemed appropriate. This procedure is currently designated as applicable only to Vogtle Units 1 and 2. Upon approval of the RICT program for FNP, the procedure will be revised to note that it is also applicable to FNP Units 1 and 2.

2.3 Calculation of RMAT and RICT for the RICT Program:

This procedure (Reference 4) provides requirements and limitations of the RICT program at SNC. It includes the guidance necessary for the calculation of RMATs and RICTs for the RICT Program. It provides the steps necessary to perform the automated calculation using the CRMP tool, as well as providing the necessary steps for a manual calculation.

For planned maintenance, personnel from the Outage and Scheduling department will calculate the RICT Program values. For emergent work, the calculation will be performed by the Operations department. If plant conditions demand that the Operations department is unable to perform the calculations, this responsibility is delegated to the Outage and Scheduling department personnel. However, entry into a Technical Specification Limiting Condition for Operation (LCO) Action statement is the responsibility of the licensed operators; this is also true for the RICT Program. Consequently, even though Outage and Scheduling may calculate a RICT in anticipation of some future entry into the RICT Program, the actual RICT will be put into place by the control room staff. In other words, the on-shift licensed operators and shift management will be generating the paperwork necessary for entry into the RICT Program, just as they do for entry into an LCO Action statement. Additionally, the Plant Manager or designee is responsible for approving entry into a planned RICT and the Shift Manager is responsible for approving entry into an emergent RICT.

The RMAT and the RICT risk levels are referenced to the Core Damage Frequency (CDF) and the Large Early Release Frequency (LERF) associated with the “zero-maintenance” state. The actual calculation evaluates the Incremental CDF (ICDF) and the Incremental LERF (ILERF) to determine the RMAT and RICT values. The evaluation is performed using the single top internal events PRA model, Fire PRA model, a bounding seismic analysis, and a bounding control room abandonment fire analysis that will be used until detailed fire modeling has been completed for these scenarios and is incorporated into the CRMP model.

The procedure contains the following guidance, restrictions and limitations, which are based on, and consistent with, NEI 06-09 (Reference 1):

- Prohibitions from entering the RICT Program voluntarily during a TS Loss of Function (LOF) Condition or when all trains or subsystems of equipment required by the TS LCO would be inoperable, unless PRA functionality has been established.
- Guidance on the use of RMAs, including the conditions under which a Common Cause Failure RMAs are developed.
- Conditions under which a RICT Program may not be used.
- States that a RICT may not go beyond the 30 day back stop limit.
- States that a RICT may not go beyond 24 hours for a Loss of Function (LOF) Condition.

- Guidance on plant configuration changes, for example, the procedure requires recalculating the RICT and RMAT within 12 hours of the change.
- Conditions for exiting the RICT Program.

The above procedural guidance is maintained in a SNC procedure. As already mentioned, the calculation of RICT Program values are the responsibility of the Operations department (emergent conditions) and the Outage and Scheduling group (planned conditions). The procedure is managed by Fleet Work Management (FWM) and is under the direction of the FWM Manager. The ownership of this procedure is subject to change if deemed appropriate. This procedure is currently designated as applicable only to Vogtle Units 1 and 2. Upon approval of the RICT program for FNP, the procedure will be revised to note that it is also applicable to FNP Units 1 and 2.

2.4 PRA Functionality Determination

This procedure (Reference 5) provides requirements for determining whether structures, systems and components (SSCs) that are declared inoperable per Technical Specifications can be considered PRA functional in RICT calculations. This procedure lists three specific conditions under which an inoperable SSC per Technical Specifications is considered “PRA Functional,” based NEI 06-09 guidance (Reference 1). They are as follows:

1) Condition 1: If the SSC is declared inoperable per Technical Specifications due to degraded performance parameters and the PRA success criteria are met, then the component may be considered PRA functional, subject to the following:

    - The degraded condition must be identified, and there is a reasonable expectation that additional degradation will not occur during the RICT.

    - For example, a valve fails its in-service testing stroke time acceptance criteria, but the response time of the valve is not relevant to the ability of the valve to provide its mitigation function as required in the PRA; therefore, the valve may be considered PRA functional.

2) Condition 2: If the condition causing the inoperability per Technical Specifications impacts one or more functions modeled in CRMP, and the inoperable SSC is still capable of supporting one or more functions modeled in CRMP, then the unaffected function(s) may be considered PRA functional.

    - For example, a valve is inoperable but secured in the closed position. Supported functions of the valve listed in a FNP RICT System Guideline require the valve to open and close. The condition can be addressed in CRMP by failing functions which require an open valve, but the valve may be considered PRA functional for functions which require a closed valve.

3) Condition 3: If the condition causing the inoperability per Technical Specifications impacts only function(s) that are not modeled in CRMP and the FNP PRA has concluded that the affected function(s) has no risk impact, then the SSC may be considered PRA functional.

    - For example, a pump backup start feature is inoperable and the feature is not credited in the PRA model (assumed failed); the RICT calculation may assume availability of the associated pump since the risk of the nonfunctional backup start feature is part of the baseline risk.

If the Functionality determination concludes that the inoperable SSC(s) is not PRA Functional, the SSC will be treated as failed during the RICT calculation.

The following additional conditions are applicable when a PRA Functionality evaluation is performed when a RICT is applied to a TS LOF Condition:

1) At least one train in a two-train system is required to be PRA Functional (for more than two-train systems, the number of trains that are required to be PRA functionality is described in Enclosure 1, Table E1-1).

2) Any SSCs credited in the PRA Functionality determination shall be the same SSCs relied upon to perform the Technical Specifications safety function, i.e., alternative SSCs cannot replace the SSCs covered by the TSs unless such SSCs have been approved by the NRC for performance of TSs safety function.

3) Design basis success criteria parameters shall be met for all design basis accident scenarios for establishing PRA Functionality during a Technical Specifications loss of function condition where a RICT is applied.

4) A 24 hour RICT backstop applies.

5) A RICT entry is not permitted, or a RICT entry made shall be exited, for any condition involving a TS loss of function if a PRA Functionality determination that reflects the plant configuration concludes that the LCO cannot be restored without placing the TS inoperable trains in an alignment which results in a loss of functional level PRA success criteria

When a situation arises requiring a “PRA Functional” assessment, site Operations department will perform the assessment and determine whether or not a specific SSC may be considered “PRA Functional.” RIE personnel will support the Operations personnel on an as-needed basis during the “PRA Functional” assessment. If the Technical Specification Front Stop will be exceeded in less than 24 hours, the formal evaluation will be performed as soon as possible.

The above guidance is maintained in SNC procedures. It is used primarily by Operations and RIE personnel with Operations personnel having the primary responsibility for making “PRA Functionality” determinations. The procedure is managed by the Administrative department and is under the direction of the FWM Manager. The ownership of this procedure is subject to change if deemed appropriate. This procedure is currently designated as applicable only to Vogtle Units 1 and 2. Upon approval of the RICT program for FNP, the procedure will be revised to note that it is also applicable to FNP Units 1 and 2.

2.5 Recording LCOs

This procedure (Reference 6) provides the Operations department with the guidance for maintaining Control Room Operator narrative logs and LCO logs as well as other control room documentation. It will be revised to address the RICT Program in a manner consistent with the existing equivalent guidance for Vogtle Units 1 and 2 (Reference 7) prior to implementation of the RICT program at FNP. The Recording LCOs procedure provides the guidance necessary for the operation of the interface tool between the operator narrative log and LCO log with the CRMP monitor.

A software interface facilitates updating CRMP when the Control Room Operators remove (or return) a component to service that affects the risk profile. The procedure provides the steps for the Operators to perform when updating their electronic narrative log and LCO log to ensure the updated status is adequately transferred to CRMP. A RICT can still be entered, managed and exited by manually entering equipment service status and LCO conditions into CRMP at the discretion of the user if the automated interfaces are unavailable, or if the user elects to not use the automatic interface capability. The Control Room Operators and the Shift Supervisor have responsibility for maintaining their respective logs. Information entry for the narrative log and LCO log (or the CRMP interface) is primarily the responsibility of the Control Room Operator at-the-controls.

The above procedural guidance is maintained as a FNP Operations departmental procedure. It is used by the FNP Operations department, and Operations management is responsible for its content and maintenance. The ownership of this procedure is subject to change if deemed appropriate.

3.0 RICT Program Training

The scope of the training for the RICT Program will include training on rules for the new TS program, CRMP modifications, TS Actions included in the program, and procedures. This training will be conducted for SNC site and corporate personnel. The personnel that will require training are as follows:

Site Personnel

- Operations Site Functional Area Manager
- Operations Personnel (Licensed and Non-Licensed)
- Operations Training
- Outage & Scheduling Site Functional Area Manager
- Outage & Scheduling Personnel
- Work Week Managers
- Nuclear Licensing Site Personnel
- Selected Maintenance Personnel
- Site Engineering
- Risk Informed Engineering Site Risk Analyst and Backups
- Other Management

Corporate Personnel

- Operations Corporate Functional Area Manager
- Outage & Scheduling Corporate Functional Area Manager
- Nuclear Licensing Corporate Functional Area Manager and Site Functional Area Manager
- Nuclear Licensing Personnel
- Risk Informed Engineering Management
- Selected Risk Informed Engineering Personnel
- Other Management

Training will be carried out in accordance with SNC training procedures and processes (e.g., Reference 8). 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. SNC has developed three levels of training for implementation of the RICT Program at Vogtle Units 1 and 2, and these will be adopted for training for implementation of the RICT Program at FNP once the FNP RICT program is approved. They are described below:

3.1 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 Technical Specifications
- New Record Keeping Requirements
- Case Studies
- Hands-on time with the CRMP monitor
  - Calculating a RMAT and RICT
- Identifying appropriate Risk Management Actions (RMA)
- Determining PRA Functionality
- Common Cause Failure Considerations

3.2 Level 2 Training

This training is geared towards Supervisors, Managers, and individual contributors who need to understand 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 monitor 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 of the Control Room Operators or the Work Week Managers.

3.3 Level 3 Training

This training will be intended for the remaining personnel who should have an awareness of the RICT Program. These employees need basic knowledge of RICT Program requirements and procedures, but they do not need working knowledge of these requirements and procedures. This training will cover RICT Program concepts that are important to disseminate throughout the organization.

All of the above training will be conducted within the procedural guidance set forth in SNC's Training and Qualification procedures (e.g., References 9 and 10).

4.0 References

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

2. Southern Nuclear Company NMP-GM-031, “On-Line Configuration Risk Management Program”

3. Southern Nuclear Company NMP-GM-031-003, “Risk Management Actions for 10 CFR 50.65(a)(4) and the Risk Informed Completion Time Program,” Version 4.0, October 2017.

4. Southern Nuclear Company NMP-GM-031-002, “Calculation of RMAT and RICT for the RICT Program,” Version 2.0, October 2017.

5. Southern Nuclear Company NMP-GM-031-004, “PRA Functionality Determination,” Version 2.0, October 2017.

6. Farley Nuclear Plant FNP-0-SOP-0.13, “Recording Limiting Conditions for Operation,” Version 34, May 2017

7. Vogtle Electric Generating Plant Units 1 and 2, 10008-C, “Recording Limiting Conditions for Operation,” Version 31, October 2017

8. Southern Nuclear Company NMP-TR-104, “SNC Training Committees,” Version 15.0

9. Southern Nuclear Company NMP-TR-415, “Systems Operator Initial and Continuing Training Program,” Version 5.0

10. Southern Nuclear Company NMP-TR-416, “Licensed Operator Continuing Training Program Administration,” Version 8.0

Joseph M. Farley Nuclear Plant - Units 1 & 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, “Risk Informed Technical Specifications Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines”

Enclosure 9

Risk and Performance Monitoring Program

Table of Contents

1.0 Introduction

This Enclosure provides summaries of the three procedures that govern the implementation of the Southern Nuclear Operating Company (SNC) Risk-Informed Completion Time (RICT) Program's Calculation of Cumulative Risk and Performance Monitoring.

Calculation of 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 Topical Report NEI 06-09, Revision 0-A (Reference 1). The Performance Monitoring Program is discussed in Section 2.3, Element 3 of Regulatory Guide (RG) 1.174 (Reference 2). Further elaboration on the Performance Monitoring Program is found in Section 3 of RG 1.177 (Reference 3). The NRC's Safety Evaluation of NEI 06-09 (Reference 1) requests that the above procedures be discussed in the License Amendment Request.

The procedures referred to are currently effective with respect to the approved Vogtle RICT program and will be made effective for FNP once the RICT program is approved for implementation at FNP.

2.0 Risk Informed Applications

This procedure contains the instructions for the calculation of cumulative risk. The Risk Informed Engineering (RIE) Department is the procedure owner and the RIE site engineer is responsible for executing the procedure. The procedure requires the calculation of cumulative risk at least every fuel cycle, not to exceed 24 months.

The procedure requires gathering historical data with respect to RICT Program entries for an assessment period which, as previously mentioned, is one fuel cycle, not to exceed 24 months. The procedure provides the method for calculating the cumulative Incremental Core Damage Probability (ICDP) and Incremental Large Early Release Probability (ILERP). These values are then converted into average annual values which are then compared to the limits of RG 1.174 (Reference 2).

If any limits are exceeded, a Condition Report (CR) is written to ensure the data is reviewed to assess the cause and to implement any necessary corrective actions to ensure future plant operation is within the guidance. This evaluation assures that RMTS program implementation meets RG 1.174 (Reference 2) guidance for small risk increases.

The procedure further instructs personnel to document the periodic assessment in a calculation including the cumulative risk, the method of monitoring the cumulative risk, comparison with RG 1.174 limits (Reference 2), and any condition reports issued including references to items that track development and/or completion of corrective actions. This procedure is under the oversight of the RIE department.

3.0 Performance Monitoring Program

Performance Monitoring is described in the Maintenance Rule implementation procedure as well as the On-Line Configuration Management procedure. This procedure is currently applicable to Vogtle Units 1 and 2. Upon approval of the RICT program for FNP, the procedure will be revised to note that it is also applicable to FNP Units 1 and 2.

The purpose of performance monitoring is to monitor the effects of the RICT on a particular SSC's performance which has had its Completion Time (CT) extended by the RICT Program. In other words, this program is used to ensure that the use of the RICT program, for a specified SSC, does not degrade the performance of that SSC over time. The SSCs in the scope of the RICT program are also in the scope of the Maintenance Rule. Additionally, it does not alter the system or train Operability requirements with respect to the number of systems and trains required to be Operable nor does it change the stated TS performance criteria (e.g. flow rate, response times, stroke times, setpoints, etc.).

These procedures are under the oversight of the Engineering Systems Department (Maintenance Rule Implementation) and Work Management (On-Line Configuration Risk Management Program). The RIE site engineer has the primary responsibility for the execution of performance monitoring program for the Risk Informed Completion Time Program. The ownership of these procedures is subject to change as deemed appropriate.

Monitoring the actual performance of a component under the Maintenance Rule is done on a monthly basis. Consequently, if it is determined that the RICT was the cause, or a contributing factor, in exceeding Maintenance Rule performance criteria, corrective actions are initiated. Although others are possible, these actions may include a moratorium on future entries into pre-planned RICTs for a period of time, or restricting the use of a RICT for specific configurations or components. Whatever the corrective actions, they are communicated to the site RIE Engineer for his or her evaluation.

4.0 References

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

2. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis,” US Nuclear Regulatory Commission, Revision 2, May 2011.

3. Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications,” US Nuclear Regulatory Commission, Revision 1, May 2011.

Farley Nuclear Plant Units 1 and 2
License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines"

Enclosure 10

Risk Management Action Examples

Table of Contents

1.0 Introduction

This enclosure describes the process for identification of Risk Management Actions (RMAs) applicable during extended Completion Times and provides examples of RMAs. RMAs for the Farley Nuclear Plant (FNP) Risk Informed Completion Time (RICT) Program are governed by an SNC fleet wide procedure. This procedure contains guidance for the determination and implementation of RMAs when entering the RICT Program and is consistent with the guidance provided in Topical Report NEI 06-09, Revision 0-A (Reference 1).

2.0 Responsibilities

The Outage and Scheduling group is responsible for developing the RMAs with support from the Risk Informed Engineering (RIE) site engineer and the Operations department on an as-needed basis. The Operations department is responsible for the implementation of RMAs. For example, if it is anticipated that a planned activity will exceed its Risk Management Action Time (RMAT), the Outage and Scheduling department will propose and develop RMAs. However, the Operations department will ultimately approve or disapprove such actions and, if approved, implement them. The same is the case for emergent activities (although in those cases it may be necessary for the Operations department to develop and implement the RMAs).

3.0 Procedural Guidance

For planned maintenance activities, implementation of RMAs will be required if it is anticipated that the RMAT will be exceeded. The RMAs are implemented at the earliest possible time, without waiting for the actual RMAT to be exceeded. For emergent activities, RMAs are implemented if the RMAT is reached. RMAs may also be required to address the potential for a Common Cause Failure (CCF) as dictated by the "extent of condition" evaluation performed subsequent to the entry of an emergent TS inoperable Condition. Additionally, if an emergent event occurs, requiring re-calculation of a RMAT already in place, the procedure requires a re-evaluation of the existing RMAs for the new plant configuration to determine if new RMAs are appropriate.

RMAs are put in place no later than the point at which an incremental core damage probability (ICDP) of 1E-6 is reached, or no later than the point at which an incremental large early release probability (ILERP) of 1E-7 is reached. Furthermore, if (as the result of an emergent event) the instantaneous core damage frequency (CDF) or the instantaneous large early release frequency (LERF) exceeds 1E-3 or 1E-4 events per year, respectively, RMAs are required to be implemented. These requirements are consistent with the guidelines provided in NEI 06-09 (Reference 1). Additionally, for emergent activities, if a high degree of confidence cannot be established that the redundant train(s) is not vulnerable to a CCF, RMAs are required to be implemented prior to the front stop being reached, specifically to address the common cause possibility. CCF RMAs to address the potential for common cause are not required if the RICT is numerically adjusted in the PRA model to account for the possibility of the common cause failure. The CCF RMAs can be discontinued if the results of the completed "extent of condition" evaluation demonstrate that the redundant train(s) is not vulnerable to a common cause failure.

The RIE site engineer, or other designated risk analyst, will provide support on an as-needed basis for determining which RMAs are appropriate for minimizing the impact of changes in core damage risk. By determining which SSCs are most important from a CDF or LERF perspective for a specific plant configuration, RMAs may be identified and implemented to protect these SSCs. Additionally, the Configuration Risk Management Program (CRMP)-generated "Remain in Service" list is an important information source for determining these important SSCs. The "Remain in Service" list provides a list of in-service SSCs that have a high impact on risk for a particular plant configuration. This listing is obtained on-demand by the CRMP user.

It is also possible to credit RMAs to affect the RICT Program calculations. However, such quantification of RMAs is not required. As stated in the procedure, omission of such a computation will result in conservative RICT Program values. However, if RMAs are to be credited, the procedure provides guidance on determining the risk impact of the RMA on RICT calculations. These include, but are not limited to, determination of RMA risk impacts on new temporary equipment functions and new or modified human actions. In addition, actions credited are required to be proceduralized and the implementing staff must be trained unless they are simple enough to be considered as skill of the craft.

4.0 Types of Risk Management Actions

Topical Report NEI 06-09 (Reference 1) classifies RMAs into three categories. These three categories are each addressed in the SNC RMA fleet-wide procedure. The fourth category is aimed at minimizing the risk of a CCF. They are described below:

1) Actions to provide increased risk awareness and control.

   A good example of this is a shift brief or a pre-job brief. Additionally, training (formal or informal) can serve to increase awareness.

   To increase control, the procedure suggests having the system engineer, or other system expert, present for the duration of the activity or certain portions of the activity. Also, a special purpose procedure may be written and used which includes the identification of the associated risk and also includes contingency plans in case of unexpected occurrences, including approaching the end of a RICT.

2) Actions to reduce the duration of maintenance activities.

   This may be accomplished by pre-staging materials, conducting training on mock-ups, performing the activity around the clock, and 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.

   The previously mentioned CRMP generated “Remain in Service” list is used to assist in determining these actions. For example, work may be stopped or minimized on safety systems redundant to the system or component being removed from service, or maintenance minimized on other systems that adversely affect the CDF or LERF.

   Minimizing work on systems that may cause a trip or transient would also be a prudent action to take to minimize the likelihood of an initiating event that the out of service component is designed to mitigate.

   Other measures that serve to minimize risk include actions like establishing temporary systems to supply power or ventilation and rescheduling or shortening other risk significant work, if possible.

4) Actions to minimize the risk of a CCF

   Many of these RMAs may be similar to those presented above but could include other types of actions that identify and possibly prevent their consequences, such as a “conditioning event” which is an event that predisposes a component to failure, or increases its susceptibility to failure, but does not itself cause failure (pump failed because of high humidity), a “trigger event” which is an event that activates a failure, or initiates the transition to the failed state, whether or not the failure is revealed at the time the trigger event occurs. An event which led to high humidity in a room in the example above, (and subsequent equipment failure) would be such a trigger event. A trigger event, in the case of CCF events, is usually an event external to the components in question and “coupling factors and mechanisms” (such factors include similarity in design, location, environment, mission and operational, maintenance, and test procedures). For example, if a bearing fails, are similar bearings used in similar circumstances or equipment that could also fail? Prohibiting switchyard work is an additional example of this type of RMA that would reduce the likelihood of an initiating event that would demand equipment redundant to the inoperable SSC. Additionally, systems redundant to the inoperable component could be protected, as well as redundant components within the same inoperable system.

5.0 Examples

The RMA procedure provides examples of types of RMAs. Examples of RMAs that are considered during a RICT Program entry are provided in the items below:

A. TS LCO: 3.8.1B (1 DG Inoperable) - Examples of RMAs that are considered during a diesel generator (DG) RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) The condition of the offsite power supply, switchyard, and the grid is evaluated prior to entering a RICT, and RMAs as identified below are implemented, particularly during times of high grid stress conditions, such as during high demand conditions;

(2) Deferral of switchyard maintenance, such as deferral of discretionary maintenance on the main, auxiliary, or startup transformers associated with the unit;

(3) Deferral of maintenance that affects the reliability of the trains associated with the operable DGs;

(4) Deferral of planned maintenance activities on station blackout mitigating systems, and treating those systems as protected equipment;

(5) Contacting the dispatcher on a periodic basis to provide information on the DG status and the power needs of the facility.

B. TS LCO: 3.8.4B (One Auxiliary Building DC electrical power subsystem with battery connection resistance not within limit inoperable) - Examples of RMAs that are considered during a safety related battery RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(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 remaining battery capacity and protect its ability to perform its safety function; and

(4) Periodically verify battery float voltage is equal to or greater than the minimum required float voltage for remaining batteries.

C. TS 3.8.1C (Two required offsite circuits inoperable) - Examples of RMAs that are considered during a two required offsite circuits RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) Limit the potential for a loss of offsite power by terminating all activities in the low voltage and high voltage switchyard.

(2) Notify the Power Control Center to defer any planned activities with the potential to generate a grid disturbance.

(3) Maintain availability of offsite power to defer any planned activities with the potential to generate a grid disturbance.

(4) Evaluate weather predictions and take appropriate actions to mitigate potential impacts of severe weather.

D. 3.8.1D (1 offsite source and 1 DG inoperable) - Examples of RMAs that are considered during a required offsite circuit and DG RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) Deferral of switchyard maintenance, such as deferral of discretionary maintenance on the main, auxiliary, or startup transformers associated with the unit.

(2) Notify the Power Control Center to defer any planned activities with the potential to generate a grid disturbance.

(3) Maintain availability of offsite power to/from A and B Startup Transformers (SUT), maintain Operability of A and B train DGs, and maintain Operability of A and B train 4160 V safety buses.

(4) Evaluate weather predictions and take appropriate actions to mitigate potential impacts of severe weather.

(5) The condition of the offsite power supply, switchyard, and the grid is evaluated prior to entering a RICT, and RMAs as identified below are implemented, particularly during times of high grid stress conditions, such as during high demand conditions.

(6) Deferral of maintenance that affects the reliability of the trains associated with the operable DGs.

(7) Deferral of planned maintenance activities on station blackout mitigating systems, and treating those systems as protected equipment.

(8) Contacting the dispatcher on a periodic basis to provide information on the DG status and the power needs of the facility.

E. TS 3.8.4A (One Auxiliary Building DC electrical power subsystem inoperable) - Examples of RMAs that are considered during a loss of a DC train RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) Limit the potential for a loss of offsite power by terminating all activities in the low voltage and high voltage switchyard.

(2) Establish 24/7 staffing and response teams to ensure prompt restoration of operability of the chargers

(3) Work to establish alternate power to the 125 V DC bus by temporary modification or by implementation of FLEX procedures

(4) Maintain Operability and availability of redundant and diverse electrical systems.

(5) Maintain/establish Operability/availability of important mitigating SSCs.

(6) Evaluate weather predictions and take appropriate actions to mitigate potential impacts of severe weather.

F. TS 3.8.1G (Automatic Load Sequencer inoperable) - Examples of RMAs that are considered during RICT for load sequencer ‘A’ so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) Limit the potential for a loss of offsite power by terminating all activities in the low voltage and high voltage switchyard.

(2) Notify the Power Control Center to defer any planned activities with the potential to generate a grid disturbance

(3) Establish 24/7 staffing and response teams to ensure prompt restoration to operability of sequencer ‘A’

(4) Perform a beginning of shift brief that focuses on actions operators will take in response to a loss of offsite power or safety injection. Include review of local emergency start of DG manual tie to 4160 VAC bus per procedure FNP-O-SOP-38.1, and manual bus loading.

(5) Maintain Operability and availability of redundant and diverse electrical systems by performing the following actions:

a. Establish protection of the following SSCs against inadvertent operation or contact that may impede the SSC from fulfilling its design function: A and B SUTs, A train DGs, B train sequencer, A and B train 4160 VAC buses; and
b. Terminate any in-progress testing or maintenance activities with the potential to impact the aforementioned SSCs; and
c. Defer any scheduled testing or maintenance activities with the potential to impact the aforementioned SSCs.

(6) Maintain/establish Operability/availability of additional important mitigating SSCs. Identify risk-significant SSCs, either from a pre-plan or by real-time use of CRMP importance reports. Perform the following actions:

a. Terminate any in-progress testing or maintenance activities with the potential to impact the availability of important in-service SSCs, and
b. Defer any scheduled testing or maintenance activities with the potential to impact important in-service SSCs,
c. Promptly return to service any important out-of-service SSCs.

G. TS 3.8.9D (One or more AC electrical distribution subsystems inoperable) - Examples of RMAs that are considered during an AC subsystem RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) Terminate any in-progress maintenance/testing activities and defer any scheduled maintenance/testing activities with the potential to cause loss of 4160 VAC safety

buses. Also, avoid unnecessary switching (e.g., breaker manipulations on ‘A’ (‘B’) train AC and DC electrical systems).

(2) Establish 24/7 staffing and response teams to ensure prompt restoration of operability of inoperable AC bus.

(3) If power cannot be readily restored through the inoperable AC bus, work to establish temporary modifications providing power to important loads fed from the inoperable bus.

(4) Maintain operability and availability of inoperable subsystem’s remaining electrical SSCs, as well as the other subsystems’ electrical SSCs.

H. TS 3.8.9E (One or more AC Vital buses inoperable) - Examples of RMAs that are considered during an AC vital subsystem RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) Limit the potential for a loss of offsite power by terminating all activities in the low voltage and high voltage switchyard.

(2) Maintain operability and availability of inoperable subsystem's remaining electrical SSCs, as well as the other subsystems' electrical SSCs.

(3) Maintain/establish Operability/availability of important mitigating SSCs.

(4) Establish 24/7 staffing and response teams to ensure prompt restoration of operability of inoperable SSC.

(5) Evaluate weather predictions and take appropriate actions to mitigate potential impacts of severe weather.

I. TS 3.8.9F (One Auxiliary Building DC electrical power distribution subsystem inoperable) - Examples of RMAs that are considered during an DC subsystem RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) Limit the potential for a loss of offsite power by terminating all activities in the low voltage and high voltage switchyard.

(2) Maintain operability and availability of redundant and diverse electrical systems.

(3) Maintain/establish Operability/availability of important mitigating SSCs.

(4) Work to establish alternate power to the 125 V DC bus by temporary modification or by implementation of FLEX procedures.

(5) Evaluate weather predictions and take appropriate actions to mitigate potential impacts of severe weather.

J. 3.8.7A (One required inverter inoperable) - Examples of RMAs that are considered during an inverter RICT, so that the increased risk is reduced and to ensure adequate defense in depth, are:

(1) Limit the potential for a loss of offsite power by terminating all activities in the low voltage and high voltage switchyard.

(2) Maintain operability and availability of DC electrical systems in the subsystem within the same train and the redundant subsystem in the other train (e.g. if the inverter in subsystem A is inoperable, maintain operability in the subsystems B and C), associated 600 V bus, and associated regulating transformer.

(3) Maintain/establish Operability/availability of important mitigating SSCs.

(4) Establish 24/7 staffing and response teams to ensure prompt restoration of operability of inoperable inverter.

(5) Evaluate weather predictions and take appropriate actions to mitigate potential impacts of severe weather.

6.0 References

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